A Comprehensive Inventory of Radiological and Nonradiological Contaminants in Waste Buried or Projected to be Buried in the Subsurface Disposal Area of the INEL RWMC During the Years 1984–2003

1. INTRODUCTION AND BACKGROUND

1.1 Objective and Overview

This report documents the compilation of a comprehensive inventory of radiological and nonradiological contaminants in waste buried or projected to be buried in the Subsurface Disposal Area (SDA) at the Radioactive Waste Management Complex (RWMC) of the Idaho National Engineering Laboratory (INEL) from 1984 through 2003. The inventory was compiled primarily for performing a future baseline risk assessment (BRA) under the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA). The project to compile the inventory is referred to as the recent and projected data task (RPDT).

This report is a follow-on to a previous report (LITCO 1995) that documents the historical data task (HDT) project. The HDT project covers waste buried at the SDA during the years 1952 through 1983. The methodologies used in the two reports are essentially identical. Taken together, the two reports encompass the waste buried or projected to be buried in the SDA from 1952 through 2003.

The inventory compiled in this document covers the waste buried from 1984 through 1993 and waste projected to be buried from 1994 through 2003. The 1984–1993 waste is referred to here as the "recent waste." At the time this study was performed, 1993 was the most recent year for which waste disposal data were available. The 1994–2003 waste is referred to here as the "projected waste" because the information is based largely on waste generators' projections, or forecasts, of waste expected to be generated in the future. At the time this study was performed, the year 2003 was the end point for the time period included in the forecasts.

In terms of disposal location, several parts of the SDA are included in the inventory. As explained in the next section of this report, the SDA consists of numerous disposal units. The disposal units covered in this task include the following:

- Pits 17 through 20 (except for the 1982 and 1983 waste that is now in Pit 17)
- The 1984 waste that is in Pits 15 and 16
- Soil Vault Rows 14 through 20
- The 1984 waste that is in Soil Vault Rows 11 and 13.

These disposal units form a completely complementary set with those addressed in the previous report (LITCO 1995). Taken together, the units cover all waste buried in the SDA.

Waste in the Transuranic Storage Area (TSA) is not included in this inventory because it is stored aboveground.

The inventory addresses radioactive waste, as well as the limited quantities of hazardous substances per CERCLA [which encompass hazardous waste per the Resource Conservation and Recovery Act (RCRA) and other hazardous substances] and mixed waste buried in the time period of interest.

Figure 1-1 presents an overview logic flowchart of the activities conducted to develop the inventory.

Several sources of information were used to compile the inventory, including process knowledge, operating logs, previous inventory-related documents, shipping records, information databases, waste generator forecasts, engineering and nuclear physics calculations, and interviews with personnel having knowledge of the facility operations that produced the waste streams.

As was the case with LITCO (1995), this task built upon the inventories in previous reports and databases by compiling several types of additional information that are needed for the BRA:

- A more comprehensive inventory of nonradiological contaminants
- Identification of specific radionuclides previously listed under generic names [e.g., mixed fission products (MFP) or mixed activation products (MAP)]
- Physical and chemical forms of the contaminants and of the host waste streams
- Uncertainties in the contaminant quantities.

To confirm its completeness, the inventory was compared with those in other reports and databases, and the reasons for any differences were explored. The list of contaminants was also compared with the list of contaminants detected in environmental monitoring conducted at the SDA.

This report is organized as follows. The remainder of this section provides a brief history and description of the SDA, discusses the regulations and regulatory agreements that create the need for the inventory information, and addresses the potential use of the inventory in other applications. The methods used to collect and compile the information are described in Section 2. Section 3 presents the resulting inventory for instances in which the contaminant quantities are known. Section 4 discusses and attempts estimates for instances in which the contaminant quantities are not known. Section 5 discusses the sources of data uncertainty, the methods used to estimate it, and the development of the upper and lower bounds. The completeness of the compiled inventory is confirmed in Section 6 by comparing it with inventories in existing reports and waste information databases, and with the environmental monitoring results.

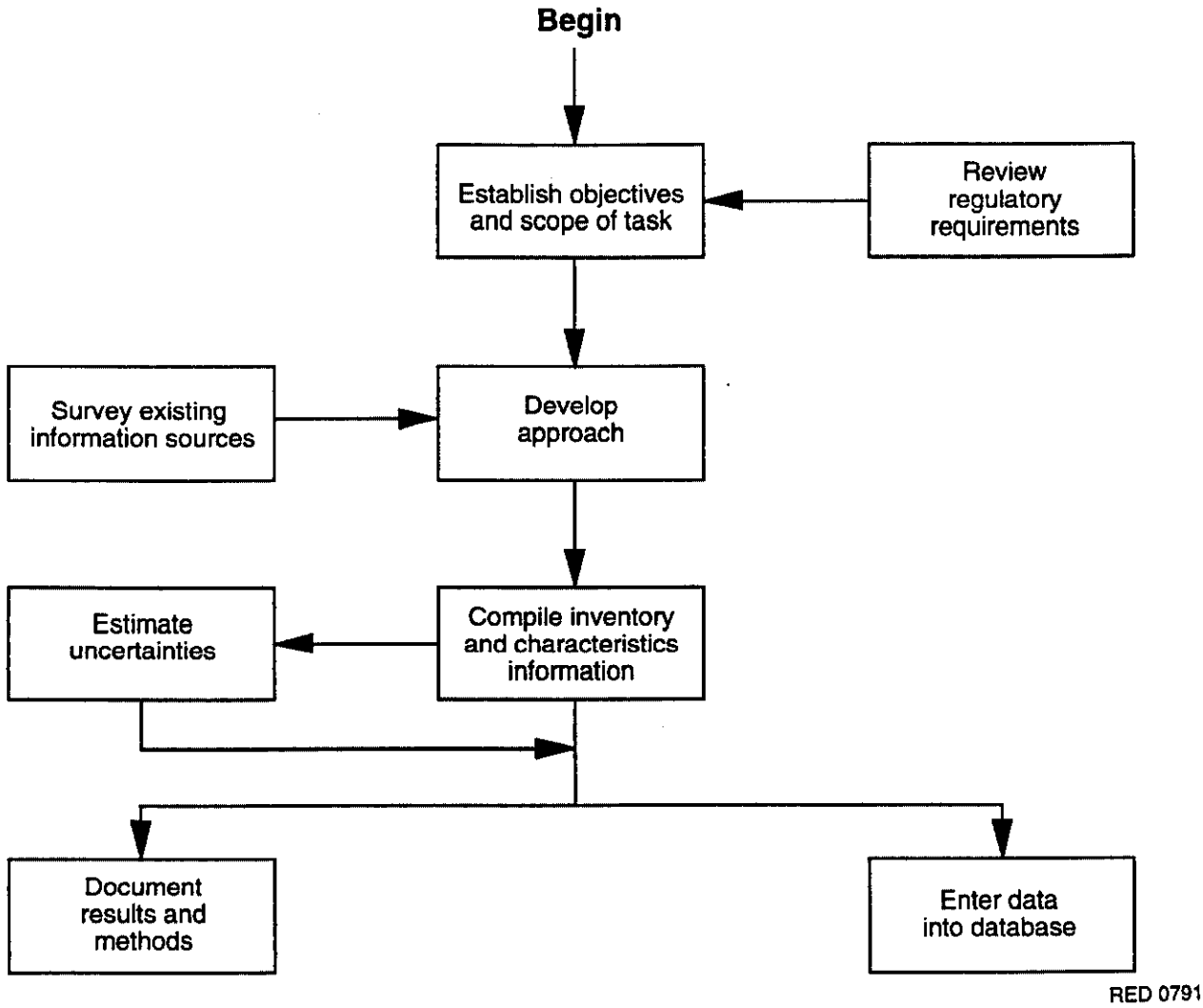

Figure 1-1. Overview logic flowchart for the task.

To the extent possible, this report is organized so as to be in parallel with the organization of the HDT report (LITCO 1995). Each section of this report contains information that corresponds to the analogous information in the corresponding section of the HDT report.

1.2 Brief History and Description of the SDA[a]

The RWMC, located in the southwest portion of the INEL, is a solid radioactive waste disposal site. The RWMC consists of the 38.85-ha SDA, the 22.7-ha TSA, and the Administrative Area (see Figure 1-2). Because the waste inventoried in this report was disposed of only in the SDA, the other two areas are mentioned only in passing.

The SDA consists primarily of three types of disposal units: pits, trenches, and soil vaults. For regulatory purposes, these disposal units are divided into various operable units, as shown in Figure 1-2.

Development of the SDA began in 1952 on a 5.3-ha tract of the original 40.5-ha site that had been identified for waste management purposes. The first shipment of radioactive waste from the INEL, which at that time was called the National Reactor Testing Station (NRTS), was buried in Trench 1 in the SDA that same year. Today, there is a total of 58 trenches; the last trench was closed in 1982.

Pits were also excavated, starting in 1957, because of the large sizes of some waste items and the increased space efficiency of pit disposal. There is a total of 20 pits in the SDA.

Containers of transuranic (TRU)-contaminated waste from the Rocky Flats Plant (RFP) in Colorado were buried at the SDA beginning in 1954 and ending in 1970. The RFP waste was interspersed with the INEL waste in pits and trenches for several years.

By 1957, the original 5.3-ha SDA was nearly filled. The SDA was then expanded eastward and southward to its present size. The expansion also enclosed the Acid Pit, which had been used since 1954 for the disposal of laboratory acids, some of which contained very low levels of radioactivity. The Acid Pit was officially closed in 1961, although records indicate that it possibly was used once in 1967 and once in 1970.

Between 1960 and 1963, the SDA also served as an interim burial ground for waste generated by licensees of the Atomic Energy Commission [AEC, a predecessor agency to the U.S. Department of Energy (DOE)]. Waste from a number of offsite generators across the country was buried at the SDA during this period. Two additional shipments of the (non-RFP) offsite waste were buried in 1967 and 1969.

---

a. This section was abridged primarily from the detailed RWMC history presented in *A History of the Radioactive Waste Management Complex at the Idaho National Engineering Laboratory* (EG&G Idaho 1985).

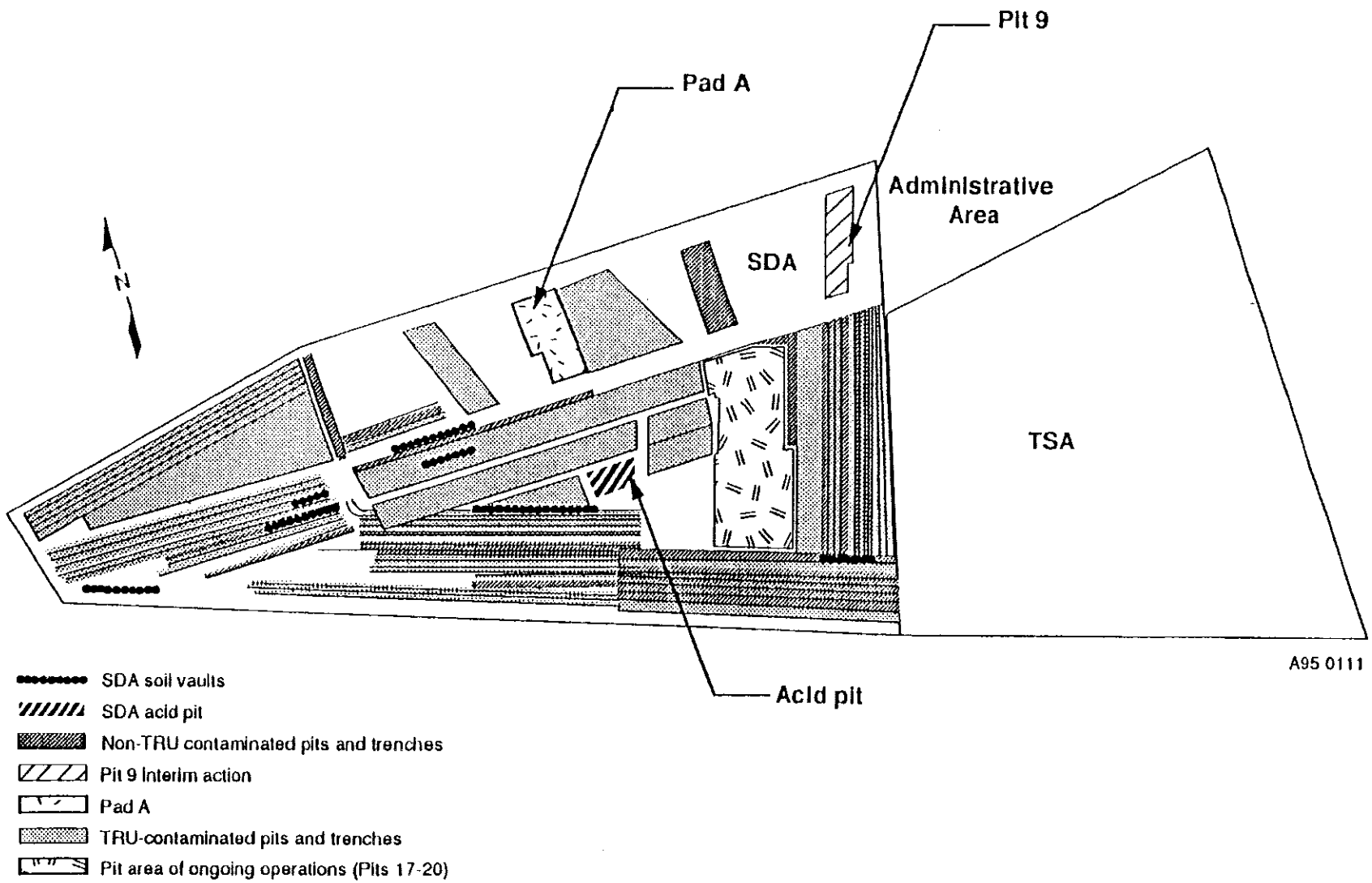

Figure 1-2. Overview layout of the Radioactive Waste Management Complex, including the Subsurface Disposal Area, Transuranic Storage Area, and Administrative Area.

Numerous changes in SDA waste management practices took place from 1952 to 1970. The general trend was toward more rigorous disposal practices. Soil-covering frequency, cover thickness, backfill over bedrock before emplacing waste, container designs, and container-stacking practices, as well as waste recordkeeping, evolved and improved over time. Several flood control and diking projects were completed, beginning in 1958 and continuing into the 1980s. Most of these projects were in response to flooding of the SDA by local runoff from snowmelt in 1962, 1969, and 1982.

In 1970, the AEC issued a policy requiring the segregation of waste contaminated with TRU radionuclides and the storage of that waste in a mode permitting later retrieval of contamination-free containers. A decision was made at the RWMC to store and cover future receipts of TRU waste[b] (and suspected TRU waste) aboveground. Accordingly, burial of such waste at the SDA ceased in 1970. Burial of non-TRU waste [low-level waste (LLW)] continues. The 22.7-ha TSA was established at the RWMC in 1970 for aboveground storage of newly received TRU waste, thereby expanding the RWMC to its present size.

In 1972, Pad A was established in the SDA for aboveground disposal of waste suspected of containing TRU radionuclides, but in concentrations less than 10 nCi/g. Pad A was closed in 1978.

Two programs demonstrated experimental retrieval of part of the waste buried in the SDA. The Initial Drum Retrieval Program (1974–1978) and the Early Waste Retrieval Program (1976–1978) retrieved approximately 150,000 ft^3 of waste, which was placed on the TSA-R storage pad in the TSA and on Pad A.

---

b. The current definitions of TRU waste and LLW are as follows, as stated in DOE Order 5820.2A:

> ***Transuranic (TRU) waste***—Without regard to source or form, waste that is contaminated with alpha-emitting transuranium radionuclides with half-lives greater than 20 years and concentrations greater than 100 nCi/g at the time of assay.
>
> ***Low-level waste***—Waste that contains radioactivity and is not classified as high-level waste, transuranic waste, or spent nuclear fuel or 11e(2) byproduct material . . . . Test specimens of fissionable material irradiated for research and development only, and not for the production of power or plutonium, may be classified as low-level waste, provided that the concentration of transuranics is less than 100 nCi/g.

Prior to 1984, the lower limit of transuranic radionuclide activity for defining TRU waste was 10 nCi/g, rather than the currently specified 100 nCi/g.

Much of the LLW and TRU waste disposed of in the SDA from 1952 through early 1984 (Nelson 1984) is ***mixed waste***: waste containing both radioactive and hazardous chemical components as defined by the Atomic Energy Act and RCRA, respectively.

In 1977, the use of soil vaults for the disposal of high-radiation-level waste began in the SDA. Soil vaults eventually replaced trenches for the disposal of such waste. The vaults are drilled in rows, as shown in Figure 1-2. As of this writing, final preparations are underway to dispose of future high-radiation-level waste in concrete vaults placed in pits.

In 1980, disposal of LLW from Argonne National Laboratory-East (ANL-E) in Illinois began at the SDA. Disposal of LLW from that generator ceased in 1988.

Beginning in 1984, DOE was required to bring all of its facilities that managed hazardous waste or mixed waste into compliance with RCRA. Acceptance of mixed waste for disposal at the SDA was discontinued in April 1984 (Nelson 1984). [An exception is contaminated lead used as shielding in waste containers, for which disposal was allowed as late as December 31, 1987 (Rodgers 1985, Rodgers 1986, Rodgers 1987).] Therefore, after that date, nonradiological contaminants identified in the RCRA regulations are not expected to be found in the waste disposed of in the SDA.

1.3 Pertinent Regulations and Agreements

This section describes the regulatory framework under which this task was performed.

Under CERCLA (or Superfund) of 1980, as amended, Federal agencies that have facilities included on the U.S. Environmental Protection Agency's (EPA's) National Priorities List are required to enter into agreements with the EPA. These interagency agreements are designed to expedite remedial actions in response to the release (actual or potential) of hazardous substances to the environment at those facilities.

On December 21, 1989, the INEL was added to the EPA's National Priorities List of Superfund sites. On December 9, 1991, a Federal Facility Agreement and Consent Order (FFA/CO) for the INEL was signed and approved by DOE, EPA, and the State of Idaho Department of Health and Welfare. The goal of this agreement is to ensure that INEL releases of hazardous substances are thoroughly investigated in accordance with the National Contingency Plan (NCP, see 40 CFR 300) and that appropriate response actions are taken as necessary to protect human health and the environment. One of the INEL waste area groups (WAGs) defined under the FFA/CO is WAG-7, the RWMC.

Under 40 CFR 300.430 (d)(2), the NCP requires the following:

> "The lead agency shall characterize the nature of and threat posed by the hazardous substances and hazardous materials and gather data necessary to assess the extent to which the release poses a threat to human health or the environment . . . ."

The RPDT and the HDT (LITCO 1995) focused on the first part of the above regulation, i.e., "characterize the nature of . . . the hazardous substances and hazardous

materials"[c] disposed of in the SDA. The BRA that this task supports will address the second portion of the requirement.

More detailed requirements concerning the characterization of hazardous substances are found in 40 CFR 300.430 (d)(2) (iii) and (iv). The information collected is to cover

> ". . . the general characteristics of the waste, including quantities, state, concentration, toxicity, propensity to bioaccumulate, persistence, and mobility" and "the extent to which the source can be adequately identified and characterized."

The RPDT addressed most of the above requirement for the 1984–2003 time period. The remainder of the requirement will be addressed in the BRA.

Guidance on complying with the NCP regulations is provided in, among other sources, the *Risk Assessment Guidance for Superfund, Volume 1, Human Health Evaluation Manual, Part A* (EPA 1989). Section 4 of that manual lists "determination of the nature of the wastes" as one of the primary data-collection components of the remedial investigation/feasibility study (RI/FS) under the NCP. Available site information must be reviewed, including "information on amounts of hazardous substances disposed (e.g., from site records)."

The RPDT was planned and conducted with close attention to the above regulations and guidance. The intent was that the resulting inventory of contaminants comply fully with all applicable requirements.

1.4 Other Uses of the Results

In addition to its use for the BRA, the inventory information has other potential uses. Although not targeted specifically for use in evaluating remedial alternatives, much of the present information may be useful for such purposes. The information collected on chemical and physical properties of the waste may be helpful in evaluating treatment alternatives, assessing health and safety hazards to workers, and identifying potential operational problems.

Caution: Other applications of this information may be appropriate only if the nature of the application is compatible with the purpose of the present study. The suitability of this information (which was developed for risk assessments under CERCLA) for use in other applications may be affected. For example, the degree of conservatism appropriate in inventory information depends on the application. For some applications, best-estimate values are appropriate. For other applications, more conservative values are appropriate. In evaluations such as safety analyses, highly conservative, upper-limit values are generally appropriate.

---

c. Generally throughout this report, the term "contaminants" is used in place of "hazardous substances and hazardous materials."

Although a major effort has been devoted to compiling this inventory, new information may be identified that could require modification of the inventory. Furthermore, some information concerning certain contaminants may never be located because of the lack of records.

References for Section 1

EG&G Idaho, 1985, *A History of the Radioactive Waste Management Complex at the Idaho National Engineering Laboratory*, WM-F1-81-003, Revision 3, EG&G Idaho, Inc., July 1985.

EPA, 1989, *Risk Assessment Guidance for Superfund, Volume I, Human Health Evaluation Manual, Part A*, interim final, EPA/540/1-89/002, U.S. Environmental Protection Agency, December 1989.

LITCO (Lockheed Idaho Technologies Company), 1995, *A Comprehensive Inventory of Radiological and Nonradiological Contaminants in Waste Buried in the Subsurface Disposal Area of the INEL RWMC During the Years 1952–1983*, INEL-95/0310, Rev. 1, formerly EGG-WM-10903, August 1995.

Nelson, J. H., 1984, letter to distribution, "Radioactive Mixed Waste," JHN-52-84, EG&G Idaho, Inc., April 24, 1984.

Rodgers, A. D., 1985, letter to distribution, "Radioactive Waste Containing Lead," ADR-23-85, EG&G Idaho, Inc., March 29, 1985.

Rodgers, A. D., 1986, letter to distribution, "Elimination of Lead from Low-Level Waste," ADR-53-86, EG&G Idaho, Inc., June 24, 1986.

Rodgers, A. D., 1987, letter to distribution, "Use of Lead in Low-Level Waste Packages Schedule Exception," ADR-46-88, EG&G Idaho, Inc., December 16, 1987.

2. METHODOLOGY FOR DATA COLLECTION AND COMPILATION

This section describes the methods by which the waste inventory information was identified, collected, compiled, reviewed, and entered into a database. Except for Section 2.4, which discusses the method for estimating contaminant quantities in future waste, the methods used in the RPDT were essentially identical to those used in the HDT (LITCO 1995).

2.1 Overview

The first step in a risk assessment is to identify and quantify all radiological and nonradiological contaminants in the waste with the potential to harm humans or the environment.

Waste disposal at the SDA began in 1952. Disposal requirements and practices at that time did not include the current requirements for waste characterization. Certainly, it was not envisioned at that time that the information provided about the waste would be used later to perform a formal risk assessment, so complete information about the waste was not obtained when it was generated and disposed of. During the 1984–1993 time period, the collection of information for newly generated waste improved gradually, although there were still some information gaps in this time period. However, as discussed later, inventory information can be and has been compiled that is sufficiently comprehensive and reliable to support a risk assessment.

One way in which contaminants are often identified is through a sampling and analysis program. Drilling, sampling, and analysis to determine an appropriate inventory for the SDA is not considered feasible or practical for several reasons: (a) the area is quite large; (b) drilling into disposal units containing radioactive waste is hazardous; and (c) the contaminants are distributed unevenly over the area, some places in concentrated form and some places in dilute form. Even a massive drilling and sampling campaign would not result in an inventory in which high confidence could be placed because of the heterogeneity of the waste.

Information and inventories concerning the waste buried at the SDA have been compiled in many previous efforts for various uses (although few of the efforts addressed waste disposed of after 1970). Some of the compilations have been entered into databases. (Sections 2.2 and 2.3 discuss existing documents and databases, respectively, that contain information on the buried waste.) Some of the compilations pertain to the entire SDA; others pertain to only one of the many disposal units. Most of the compilations were derived from shipping records. (Section 2.3 discusses some of the deficiencies in the shipping records.) Many of the inventory compilation efforts addressed only the radioactive component of the waste. Further, waste information obtained for one purpose often does not provide all of the parameters needed when used for a different purpose. After investigation, it was concluded that the existing compilations of waste inventory information were very useful, but not adequate to support a risk assessment of the SDA under CERCLA.

In view of the limitations of the above approaches, a different approach was devised to collect the information. The approach emphasized the use of process knowledge.

First, the facilities that generated or that are expected to generate the SDA waste during 1984–2003 were divided into six groups, as follows:

- Test Area North (TAN)
- Test Reactor Area (TRA)
- Idaho Chemical Processing Plant (ICPP)
- Naval Reactors Facility (NRF)
- Argonne National Laboratory—West (ANL-W)
- Other generators — this covers all other onsite facilities (including environmental restoration activities and decontamination and decommissioning [D&D] programs as waste generators at those facilities) and all offsite facilities.

In the HDT report (LITCO 1995), RFP was another major waste generator. However, no waste from RFP was buried in the SDA during the time period covered in this study.

Figure 2-1 shows the geographic locations of the major and smaller waste generators at the INEL. In addition, waste with a very small amount of radioactivity (<1 Ci) is projected to be generated at the INEL Research Center in Idaho Falls. The one offsite generator is discussed in Section 2.5.6.

Six lead data gatherers were appointed to compile information on the waste from the six generators. In nearly every case, each lead data gatherer either was working at or had previously worked at the waste generator location whose information he was assigned to collect, and was familiar with the operational activities that generated the waste. Thus, the approach was *primarily one of evaluating the waste based (where possible) on knowledge of the specific processes that generated it*, as well as on review of pertinent records, databases, forecasts, and reports, rather than on simply rereviewing the shipping records.

Figure 2-2 depicts the flow of information in this approach. The rectangles represent items of information, and the ovals represent technical activities performed on the information. Section 2 describes the data-gathering activities depicted in Figure 2-2.

The upper left portion of the figure shows the principal sources of information used by the data gatherers. The data gatherers used process knowledge and plant operating records, inventory and other reports, shipping records (and databases of such records), interviews with plant employees (including retired employees), engineering and nuclear physics calculations, waste generator forecasts, and other records.

The question next arose about the level at which the waste should be characterized. The goal was to divide a given generator's waste, for data-gathering purposes, so that the resulting information would be organized in the most useful way for application to the risk assessment.

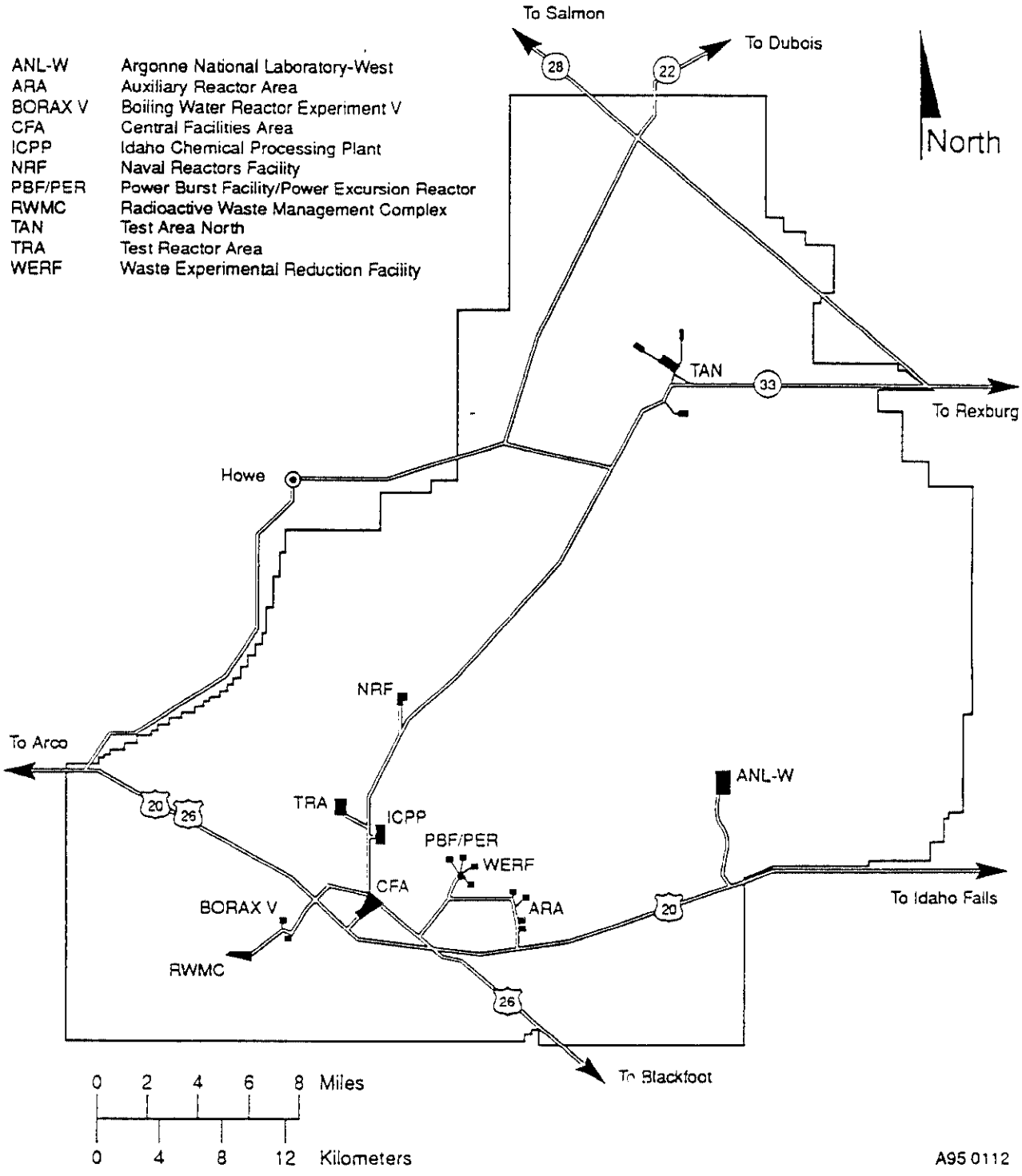

Figure 2-1. Locations of the Idaho National Engineering Laboratory waste generators in 1984–2003 and the location of the Radioactive Waste Management Complex.

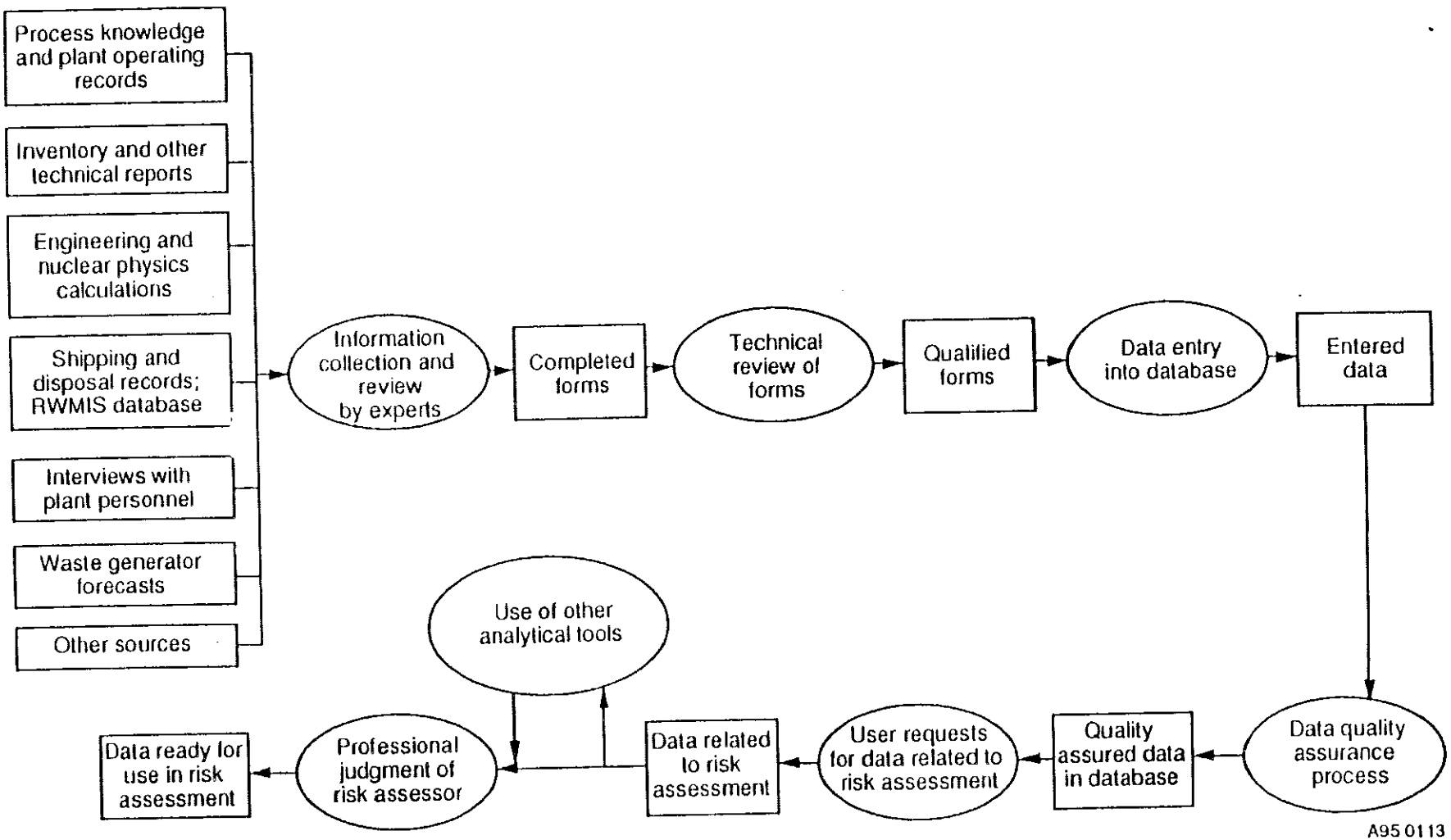

Figure 2-2. Approach for information flow in developing the inventory.

Characterization of waste at the generator level would not provide sufficient detail because the waste from a given generator varied greatly in form, constituents, and characteristics. Characterizing each waste container individually was obviously not feasible. Even if information were available on the contents of each waste container (which it is not), the number of containers involved is in the hundreds of thousands.

The approach used was to divide the waste from a given generator into "waste streams." (Dividing the waste into streams was strictly for purposes of convenience in organizing the data and did not in any way restrict the data that could be gathered.) Although the definition used herein for a waste stream is rather flexible, the term generally refers to a collection of waste containers whose contents are similar. In some cases, streams could be defined that were fairly uniform from one container to another. For example, all of the beryllium reflectors from TRA were defined as one waste stream. On the other hand, for a minor building that produced a very small amount of assorted waste, all waste from the building was generally lumped together into one, nonuniform stream.

Applying this approach led to dividing the waste from a given major generator into anywhere from 1 to 15 waste streams for either the recent or projected period. The total number of streams was 99, a manageable number, for the recent plus projected waste.

A standardized, five-page data form (see Appendix A) was used to record the information collected for each waste stream. The form indicates the generator, building, and assigned number of the waste stream from that building; the volume, physical and chemical form, and containment of the waste stream itself; the quantities (including uncertainties) and physical and chemical form of the nonradiological contaminants and radiological contaminants in the stream; the source(s) and reliability of the information; and the assumptions made in dealing with the stream. A five-page form (plus continuation pages as needed) was completed for each of the 99 streams that were identified.

Many of the information items on the data forms were established as computer-searchable data fields with prescribed lists of possible answers. However, to allow the broadest flexibility in describing the waste, several "free" fields were included on the forms where verbal descriptions could be entered to whatever level of detail was needed. Although such free fields cannot be rolled up using the database, some of the information in the free fields is invaluable in understanding subtle characteristics of the waste that affect parameters such as the mobility of the contaminants.

Candidate nonradiological and radiological contaminants for Parts C and D of the data forms (see Appendix A) were addressed as follows.

All radionuclides identified in the waste streams were included on Part D. The radionuclide K-40 was excluded from the inventory, however, based on the following rationale. The database of shipping records for 1984–1993 includes several K-40 entries in waste from several generators, totaling 0.17 Ci. No K-40 entries were identified in the waste generator forecasts. Except for the generation of K-40 by the activation of potassium used in specialized nuclear reactors, no credible mechanism could be identified for the production of K-40 in the waste. Therefore, the K-40

entries in the shipping records are believed to be spurious, the result of attributing natural background radiation from K-40 to radiation from K-40 in the waste. One possible exception might be certain waste from ANL-W, where potassium is used in nuclear reactors; however, the ANL-W data gatherer indicated that there was no K-40 in the waste from ANL-W.

K-40 is a very long-lived (half-life of 1.28 billion years) radionuclide. It occurs naturally worldwide, constituting 0.0117% of the potassium in the earth's crust (General Electric Company 1989). K-40 also is found naturally in human bodies worldwide in sufficient concentrations to produce radiation doses of 14 to 27 mrem/yr to various organs (NCRPM 1987). The average concentration of naturally occurring K-40 in INEL surface and near-surface sediments is 19 pCi/g (Hoff et al. 1993). These sediments are used for backfill in the SDA. The amount of naturally occurring K-40 in the backfill in Pits 17–20, where most of the waste addressed in this document has been or will be buried, was calculated to be approximately 5.2 Ci. This value is considerably larger than the 0.17 Ci of K-40 listed in shipping records as being in the waste. Because the total radioactivity of K-40 in the backfill is much larger than that listed in the shipping records and because the K-40 entries are believed to be spurious, K-40 was not included in the inventory compiled in this document.

Candidate nonradiological contaminants for Part C were addressed by screening against two lists. One list consisted of the hazardous substances designated by the EPA under CERCLA. The list included chemicals designated under the Federal Water Pollution Control Act, Solid Waste Disposal Act, Clean Air Act, Clean Water Act, RCRA, and Toxic Substances Control Act. The second list covered contaminants listed in the National Primary Drinking Water Standards of the Safe Drinking Water Act. If there was any question about whether to include a nonradiological contaminant, it was included. One class of nonradiological contaminants not included on Part C was metals commonly found in alloy form in structural components, i.e., nickel and chromium, which are used in stainless steel. A literature review and analysis (Weidner 1993) indicated that, considering the extremely slow corrosion rate of stainless steel in the RWMC soil and the very limited solubility of nickel and chromium at the pH of interest, the mobility of these chemicals under these circumstances is expected to be extremely limited.

The data were collected separately for the two time periods of interest (1984–1993 and 1994–2003) because the collection methods differed somewhat. The method for the recent waste (1984–1993) was identical to that for the 1952–1983 waste (LITCO 1995) and is discussed throughout Section 2. The special provisions required to address the projected waste (1994–2003) are discussed in Section 2.4.

The steps in Figure 2-2 are discussed in more detail in the remainder of Section 2. The use of source documents is discussed in Section 2.2. Section 2.3 describes the use of an existing database of shipping and disposal records. The method for estimating contaminant quantities in future waste is described in Section 2.4. Section 2.5 provides a detailed description of how the waste information was obtained for each waste generator.

After the information was collected and entered onto data forms, it was subjected to a qualification process (discussed in Section 2.6) and entered into a contaminant inventory database for risk assessment (described in Section 2.7). The database for the information gathered in the

current study is the same database used for the 1952–1983 information (LITCO 1995). Finally, with the use of other analytical tools and the professional judgment of risk assessors, the data are ready for use in risk assessments.

2.2 Use of Source Documents

As indicated in Figure 2-2, technical reports and other documents containing inventory and related information about the waste buried or projected to be buried in the SDA were one of the primary sources of information collected in the current study. This section discusses the types of reports available and describes how the reports were used.

A large number of documents contain useful information about the waste buried or projected to be buried in the SDA. The types of documents range from brief letters to comprehensive technical reports. In scope, the documents range from narrow (addressing only one waste stream from one generator) to comprehensive (fairly complete inventories), although none of the documents cover the full scope required for the BRA. Some of the documents are devoted solely to discussions of inventory, while others address inventory only briefly as part of another topic, such as the characteristics of waste to be processed in a proposed treatment facility. Many of the documents contain data extracted from previous documents. The dates of the documents range from the 1950s to the present. (Because this study addresses waste buried starting in 1984, most of the documents that were used date from 1984 onward.) Some of the documents offer crucial information, while others are of limited value.

Because the existing documents were of considerable value to the current study, as many as possible were identified and evaluated for their applicability. Data gatherers reviewed the documents related to their assigned generator and incorporated the information, as appropriate, into the present data-gathering effort.

One type of document of particular use for the projected waste was the waste generator forecasts. These forecasts are discussed in Section 2.4. Program planning documents were also useful for the projected waste.

For each waste stream, the data gatherer specified the sources of information on Part E of the data form (see Appendix A). If a document was the source of an item of information, the box titled "reports" was marked on the data form, and the title, author, report number, and date were entered. On many data forms, the data gatherers also compared the inventory specified in a reference report against sometimes-conflicting data from other sources of information, made a judgment as to which data were considered more credible, and indicated the basis for the judgment.

Approximately 60 specific reports and letters are referenced and discussed in later parts of Section 2.

2.3 Use of the Radioactive Waste Management Information System

In addition to process knowledge, technical reports, shipping records, and interviews, existing databases were searched to obtain information. The principal database accessed in the current task was the Radioactive Waste Management Information System (RWMIS).

2.3.1 Description of RWMIS

RWMIS (Litteer 1988) is a mainframe electronic database developed in 1971 that resides on an IBM 3083 computer. Information reported in RWMIS includes all airborne (onsite effluent), liquid, and solid radioactive waste shipped to or generated at the INEL. The purpose of RWMIS is to provide an inventory of radioactive waste stored or disposed of at the RWMC and radioactive effluents generated at the INEL.

The data in RWMIS originated from shipping and disposal forms[a] that accompanied the waste when it was shipped for storage or disposal.

The database consists of *summary* waste shipping and disposal records for the years 1954 to 1970 (nothing from 1952 to 1954), *waste shipment* records for 1971 to 1986, and *container-by-container* records from 1986 to the present. Shipment-specific waste information data prior to 1971 are not included in the user-accessible database. These pre-1971 records are referred to as the Best Available Data (BAD) database. Records in RWMIS for 1971 to 1983, and 1984 to the present, are referred to as the historical database and the current database, respectively.

RWMIS is a hierarchical database consisting of a parent-master (shipment information) and two children: a nuclide information child and a container information child. The parent-master has a one-to-many relationship with the nuclide and container information children.

The primary fields in RWMIS and a brief description of each are provided in Table 2-1.

Verification of the data entered into RWMIS is performed according to a process described in Litteer (1988).

2.3.2 RWMIS Download for the Current Task

A download of the current mainframe electronic databases from NOMAD to an IBM personal computer dBASE environment was performed to support the RPDT. The download was performed for the years 1984–1993.

To facilitate the use of RWMIS in the dBASE environment, the data were downloaded into three relational databases. These databases consisted of the master (as stored in RWMIS), a nuclide information database (with key information from the master), and a container information database (with key information from the master). A verification procedure was written and

---

a. For simplicity, shipping and disposal forms are generally referred to herein as "shipping records."

Table 2-1. Primary information fields in the Radioactive Waste Management Information System database.

Primary RWMIS fields	Description
Waste origin	The site (area and building/location) at which the waste was generated.
Waste type	The physical phase of the waste (i.e., liquid or solid).
Radioactive	A flag that specifies whether or not the shipment is radioactive.
Report date	The date generally identifies the date that the shipment form was completed. It usually appears on the form as the date of approval for shipment of the waste.
Waste description	A generic description of the shipment. In most cases, this field also includes radiation readings taken at contact and at 1 m from the shipment.
Gross volume	The gross volume of the waste shipment in cubic meters.
Gross weight	The gross weight of the waste shipment in grams.
Gross curies	The gross curies in the waste shipment.
Disposal date	The date of waste disposal or storage.
Container type	The type of waste container.
Container number	The number of waste containers in the waste shipment.
Container volume	The volume of each type of waste container in the waste shipment.
Volume unit	The unit of volume for each waste type container.
Waste description	The content code that provides a generic description of the waste in the container (e.g., Code 003 implies paper, metal, and wood).
Disposal location	The disposal or storage location of the waste.
Nuclide	The isotopic nuclide designation.
Nuclide weight in grams	The gram quantity of each nuclide in the waste shipment.
Nuclide quantity	The curie quantity of each nuclide in the waste shipment.

implemented to ensure that the integrity of the RWMIS databases was maintained during the download. In the RWMIS mainframe environment, a count was made of the number of records, and all numerical fields were summed. The same checks were made on the download (dBASE) version of the database. All inconsistencies were resolved before the data were used.

An application was developed to display the RWMIS data in a convenient form, called a technical note, for the RPDT.

The download version of RWMIS was used as one source of information to support the RPDT.

2.3.3 RWMIS Limitations

Section 2.1 indicated that existing compilations of SDA waste inventory information—both reports and databases—were very useful, but not adequate to support risk assessments under the FFA/CO. This section provides more detail on why the RWMIS database could not serve as the sole source of inventory information.

A primary limitation of RWMIS is that it contains very little information concerning nonradiological contaminants in the waste.

Prior to 1986, RWMIS stored data only on a shipment basis. The curies (or grams) identified with each isotope are specified for the entire shipment and not for individual containers. This limitation makes it difficult, for example, to determine if the contents of an individual container should be classified as TRU waste or LLW.

Although post-1986 data are available in RWMIS on a shipment basis, the current data form does not accept textual descriptions of the waste, but rather content codes only. For waste that does not fall conveniently within a content code, certain detailed descriptions previously available in text may not be captured electronically.

From a risk assessment perspective, there are several other deficiencies in the RWMIS database, although some of the deficiencies have been remedied for waste generated recently. The deficiencies reflect a lack of either detail or completeness. Some of the deficiencies are:

- Entries with only one radionuclide identified, e.g., Pu-239, whereas knowledge of the waste-generating process indicates that other radionuclides must also be present
- Entries with only the element specified, e.g., uranium, with no designation of a particular radionuclide
- Entries with only MAP or MFP identified, with no designation of particular radionuclides
- Entries with equal amounts of MAP or MFP identified, suggesting that no rigorous estimate of radionuclide breakdown was performed

- Entries with only one fission product identified, e.g., Cs-137, whereas knowledge of the waste-generating process indicates that others must also be present
- Entries with only one activation product identified, e.g., Co-60, whereas others must also be present
- Entries with unidentified radionuclides, e.g., unidentified beta-gamma, unidentified alpha
- Entries with no chemical form specified
- Entries with no physical form specified
- Entries that are actually for temporarily stored waste.

2.4 Method for Estimating Contaminant Quantities in Future Waste

2.4.1 Background

The inventory addressed in this study includes waste expected to be generated in the future, as well as waste already generated. The quantities and characteristics (e.g., contaminant concentrations) of waste that will be generated in the future cannot be predicted with certainty. Unforeseeable changes in many factors could substantially affect the quantities and characteristics of the waste that will be generated. Examples are changes in program funding, changes in the regulations that govern waste management, changes in facility operations, and unpredicted outcomes of decisions made in environmental restoration (ER) programs that generate waste.

To estimate the quantities of contaminants in future waste, therefore, requires that numerous assumptions be made. The assumptions are typically based on patterns of waste generation in the past, recent trends, and projections of future funding levels and facility operations.

2.4.2 Available Projections of Future Waste

Information on projected contaminants in future waste is somewhat limited.

Since 1977, near the end of each calendar year, the INEL waste generators have forecasted their future generation of waste. The waste generator forecasts consist of approximately 20 sets of forms completed by the various generators. The forecasts cover the upcoming year and the nine following years. Revisions to individual generator forecasts are sometimes provided between the annual cycles.

The most recent generator forecasts available when this task was performed cover the waste expected to be generated in calendar years 1994 and 1995, by quarter, and in calendar years 1996 through 2003. The generator forecasts include information on the type, volume, total

radioactivity, and radionuclide distribution for each "waste stream characterization number." Approximately 170 such numbers are in use. The radionuclide distributions supplied in the generator forecasts range from very simple (one radionuclide) to quite detailed (about 20 radionuclides). For some streams, particularly streams generated by D&D, no radioactivity total or distribution was supplied by the generator. No generator forecast information was provided from some of the ER WAGs.

The generator forecasts are compiled by RWMIS personnel in an annual letter (e.g., Randall 1994). The compilations are rollups over all waste streams from a given generator. Also, the compilations do not include radionuclide distributions. Any generator forecasts with "unknown" entries for total radioactivity appear as zero radioactivity on the rollups, even though the waste volume appears as a non-zero value.

Neither the generator forecasts nor the compilation provides any information on the nonradiological contaminants in the waste to be disposed of in the SDA. (Nonradiological contaminants in waste disposed of in the SDA were greatly reduced starting in 1984.)

Another compilation of projections for future waste is given in a report produced semiannually covering the ER and D&D activities (e.g., DOE 1993 and its subsequent updates). Again, no information is presented for some of the WAGs. The report provides more detailed information on the source, type, and management plans for the waste than does the annual compilation letter described above. However, it does not provide any information on radiological or nonradiological contaminants in the waste to be disposed of at the SDA. Also, the projections do not extend very far into the future. No waste from ER or D&D is projected in the report to be disposed of in the SDA after 2001, although D&D may generate a considerable amount of waste requiring disposal in the SDA after 2001.

2.4.3 Accuracy of Past Projections

The radioactivity values in past generator forecasts were compared in detail against those listed when the waste was actually disposed of at the SDA. The comparisons reveal that, although the accuracy of the projections has improved somewhat over the years, it is still limited.

An early version of the *Waste Management Plan, Annual Update* (EG&G Idaho 1988) evaluated the accuracy of the generator forecasts for waste disposed of from 1978 through 1986. In 6 of the 9 years, the forecasts overestimated the future annual amounts of radioactivity in the waste by anywhere from a factor of 2 to a factor of 7. In 3 of the 9 years, the forecasts proved to be too low. Those differences ranged from about 10% to a factor of 5.

For the current study, the radioactivity values in generator forecasts made in the four most recent years were compared with those listed when the waste was actually disposed of. The detailed statistical analysis is presented in Section 5.5.2. The conclusion is that, on the whole, the forecasts of radioactivity are biased upward by a factor of approximately 4. There is no reason to believe that the bias exhibited in the recent past has changed. Therefore, the same bias was assumed to affect the projections that were made in late 1993 in Randall (1994).

2.4.4 Waste-Projection Approach Selected for the Present Study

The waste generator forecasts, the most recent of which extend generally through 2003, were used as the starting point for projecting the waste characteristics for 1994–2003. The forecasts are considered the best data available and are the official projections for the INEL. (As discussed in Section 2.4.3, the recent waste generator forecasts of radiological contaminants have exhibited an overall upward bias of a factor of about 4. The upward bias and some other limitations of the forecasts were adjusted for as discussed below.)

The approach for the projected waste was as follows:

- A set of waste streams was developed in the current study for waste disposed of in the past years, 1984–1993 (the recent period).

- The data gatherers discussed the projected 1994–2003 waste with the person(s) who prepared the forecasts for their respective generators. In some cases, the data gatherers were the same people who had prepared the generator forecasts. The data gatherers also reviewed any program planning or other documents that could provide useful information for projecting waste generation.

- The 1993 generator forecasts for the period 1994–2003, by waste stream, were compared against the actual waste streams for 1984–1993. For each major generator, a decision was made by the data gatherer on whether to use the waste stream organization in the generator forecasts or that for the 1984–1993 waste, or a combination of the two. If the current study identified additional streams in the 1994–2003 period (not listed in the generator forecasts), that information was also factored in. The result of this step was a set of waste streams to use for 1994–2003.

- Unless there was a valid reason to change them, the generator forecasts of *total* annual radioactivity in each waste stream in 1994–2003 were used as a starting point. The upward bias was corrected through multiplication by a factor of 0.25 if the generator forecasts were used directly to project the total radioactivity. (In some special cases, waste with well-characterized radioactivity has already been generated, but is still awaiting disposal. The factor of 0.25 was not applied in such situations.) The radioactivity in the ER-generated waste for which information was not submitted in the annual generator forecast—WAG 8 (NRF) and WAG 9 (ANL-W)—was estimated and added in.

- By the use of methods appropriate to the generator and the waste stream, (e.g., nuclear physics calculations, discussions with generator technical personnel), the generator's radionuclide *distributions* for each waste stream were modified as necessary. For example, adjustments to the distribution of U-234, U-235, and U-238 were made as described in Section 3.1.

- A correction was applied to any waste radioactivity projections that had been based on extrapolations of data derived from the Geiger-Müller (G-M) counter survey method. The bias of this method, the derivation of the correction, and the application of the correction to past waste and projected waste are described in Section 5.

The generator forecasts do not address nonradiological contaminants in the waste. In this study, the projections of nonradiological contaminants are based on a combination of information sources: program planning documents, interviews, and extrapolations from corresponding waste streams of the 1984–1993 time period.

2.4.5 Volumetric Disposal Capacity of the SDA

The volume of waste that can be disposed of in the SDA is limited. The limited remaining space available for disposal of future waste has been a subject of concern for several years. Periodic studies have been performed to project the remaining lifetime so that adequate disposal capability will be ensured.

Because of this limitation, a check was made to determine whether all of the waste projected to be generated through the year 2003 could be accommodated in the SDA. The most recently published projections (DOE 1994a) indicate that initial closure of the SDA could begin as early as the year 2004 and be completed by 2007. However, ongoing studies are investigating means of extending the lifetime of the SDA well beyond those dates. The conclusion is that the SDA can be expected to accommodate all of the waste projected to require disposal through at least the year 2003. Therefore, questions about the future closure of the SDA are not expected to impact the present study.

2.5 Data Collection Methods

This section discusses the methods used to collect the waste information for each of the six waste generators. Because the waste and the available information differed from one generator to another, and even from one stream to another, the data-collection methods also differed from stream to stream.

The discussion of the methods is presented in two ways, as described below.

First, Sections 2.5.1 through 2.5.6 describe, in general terms, the waste generator of interest, the processes by which the waste was generated, the availability of information for the waste from that generator, and the data-collection approach selected.

Second, information on the assumptions and the sources of information for every waste stream is available on the data forms for the various streams. As discussed in Section 2.7, the data forms have been entered into a database. A printout of the entire contents of the database is provided in Appendix B, Volumes 2 and 3 of this report.

An alphanumeric designator is used to uniquely identify each waste stream. The first part of the designator is a (generally) three-letter code representing the name of the major generator. The second part is a three-digit code representing the building number where all or most of the stream originated.

The third part is a number representing the sequence of the waste stream identified from the given building. A suffix is added to the end of the waste stream number to indicate if the stream is recent (R) or projected (P). Thus, the designator TRA-603-21R represents the 21st waste stream identified and characterized from Building 603 at TRA; the stream was produced in the recent time period.

The stream numbers for the current study are coordinated with those from the study of the 1952–1983 waste (LITCO 1995) in a common database. For example, suppose that building TRA-601 produced three waste streams (TRA-601-1, TRA-601-2, and TRA-601-3) during 1952–1983. If building TRA-601 produced a new waste stream in the recent period, it would be designated TRA-601-4R. If, on the other hand, stream TRA-601-1 continued in the recent period, it would be designated TRA-601-1R. If stream TRA-601-2 was no longer generated in the recent period, no entry for TRA-601-2R would appear. Thus, some gaps appear in the stream numbers during the recent and projected periods.

Information about waste produced at the various generators before 1984 is available in a previous companion report (LITCO 1995).

2.5.1 Test Area North

The Generator. TAN lies at the north end of the INEL, about 27 mi northeast of the Central Facilities Area (CFA) (see Figure 2-1). TAN was designed and constructed in the early 1950s to support the General Electric Aircraft Nuclear Propulsion (ANP) Program, the mission of which was to test the concept of the nuclear-powered airplane. For a 9-year period, until the program was canceled by the U.S. Congress in 1961, the program was involved in the testing of three versions of a full-scale, nuclear-powered aircraft engine (Wilks 1962). The program support facilities consisted of the Technical Support Facility (TSF), where technical support facility personnel were officed; the Initial Engine Test (IET) Facility; the hot shop, a large hot cell into which the engines could be moved for repair, assembly, and disassembly; and some smaller hot cells, built for the examination of individual irradiated fuel pieces or other irradiated specimens. The IET and hot shop were connected by a double set of rail tracks that allowed the engines to be moved back and forth.

Also located in the TAN area, west of the TSF, was the Loss of Fluid Test (LOFT) Facility, a scaled-down version of a nuclear-powered utility generating station. The Mobile Test Assembly (MTA), which contained the LOFT reactor vessel and the primary coolant system, was mounted on a flat car that could be moved on the double set of rail tracks to the TAN hot shop for repair and disassembly. This facility was used from 1978 through 1985 by the U.S. Nuclear Regulatory Commission (NRC) to conduct tests to determine the validity of nuclear safety codes and to identify critical parameters for accident scenarios. The last two LOFT tests, LP-FP-1 and LP-FP-2, conducted under the auspices of the Organization for Economic Cooperation and Development (OECD) in 1984 and 1985, were accident scenarios to determine the fate of fission products during and after a fuel-damaging accident. The last test, LP-FP-2, was to simulate, to a limited extent, the conditions that existed in the Three Mile Island (TMI) Unit 2 generating station in March 1979.

The ANP hangar building, TAN-629, was occupied by the Specific Manufacturing Capability (SMC) project starting in 1985. The project produced armor plate for U.S. Army tanks.

After the termination of the ANP Program in 1961, the TAN facilities were used to support the LOFT Program and many miscellaneous projects. In 1984, the commencement of this study period, the LOFT Program was preparing to conduct the LP-FP-1 test; the IET Facility was shut down, with no operations occurring there; and the TAN hot shop was receiving the EPICOR and Submerged Demineralizer System liners from the TMI plant. These vessels, filled with various organic and inorganic resins, had been used to decontaminate the 600,000 gal of highly contaminated water that had leaked to the reactor building basement from the TMI primary coolant system. The TSF supported the LOFT Program and the TMI Support Program. Other facilities (TAN-603, TAN-623, and TAN-653) also supported these two programs. TAN-629 was used as a storage facility in 1984. In the subsequent 10-year period, from 1984 through 1993, various TAN facilities supported the SMC, the TMI fuel/fuel debris receipt and storage for DOE, the Prototypical Consolidation Demonstration Program, the Dry Fuel Cask Storage Program, and many smaller projects.

The activities projected to occur at TAN during the period 1994–2003 are uncertain. However, the TMI fuel/fuel debris and other spent fuel currently stored in the TAN pool is projected to be moved into dry storage at ICPP over the next 6 or 7 years (DOE 1994b). Waste to be generated during this operation will be disposed of at the RWMC (DOE 1994b). Also, SMC will generate some waste.

Generation of the Waste. Most of the waste produced at TAN during the period 1984–1993 was a result of the specific test evaluation and production programs just described.

LOFT conducted the LP-FP-1 test in 1984 (Adams 1985) and the LP-FP-2 test in July 1985 (Adams et al. 1985; Carboneau et al. 1987). Following the LP-FP-2 test, which released a significant amount of fission products from the fuel to the primary coolant system (PCS), the LOFT containment building was stripped of all hardware. The material removed was sent to the RWMC as waste. The rail flat car-mounted MTA, which contained the LOFT reactor vessel and the PCS, had the components removed, including the reactor vessel, and they were sent to the RWMC as waste.

The SMC process used depleted uranium and created a waste stream containing principally sand-blast grit contaminated with depleted uranium.

Operations performed in the hot cells and hot shop complex supported the packaging of TMI abnormal waste and the grouting of Submerged Demineralizer System resin liners into high integrity containers. These operations, which created the most LLW, began just before 1984 and were completed in 1990.

In 1984, the highly contaminated TMI control rod drive screws were received in the hot shop. In addition to the principal operations of grouting of the TMI liner resins, work supporting the LOFT FP-1 and FP-2 center fuel module examinations was performed in the hot cells and hot

shop. In 1985, Virginia Electric Power Co. fuel, H. B. Robinson fuel, and Peach Bottom fuel were transferred from the shipping casks to dry storage casks. In 1986, the Dry Rod Consolidation Technology Project was demonstrated in the TAN hot cells. In mid-1987, the TMI fuel/fuel debris shipments began to arrive at the TAN hot shop for storage in the TAN pool. This fuel receipt and storage operation continued until mid-1990. All of these activities resulted in the generation of waste.

Since the activities at TAN during 1994–2003 are uncertain, but are expected to be minor in scope, generation of future waste at TAN for disposal at the RWMC is also expected to be minor.

General Availability of Information. TAN waste information is available in RWMIS. TAN hot shop and TAN hot cell logs for the time period are available from the Federal repository and were reviewed. Specific project reports and letters were also sources of information.

For the period 1994–2003, minimal information is available in the waste generator forecasts and a limited number of programmatic documents.

Data-Collection Approach. The general data-collection approach used for the period 1984–1993 involved reviewing the information in RWMIS, facility operating logs, and specific project reports, and interviewing plant personnel.

Some minor incongruities were identified in the record for waste generated at TAN. For example, TAN building designations were not always adhered to when designating where the waste was generated (e.g., LOF-647 was used instead of TAN-647).

Another incongruity at the beginning of the recent period was that, occasionally, a particular facility would send waste to the RWMC that was generated at another facility but labeled as waste from that particular facility. For example, TAN-607 was sending waste from the Water Reactor Research Test Facility (WRRTF) and other locations to the RWMC labeled as TAN-607 waste (Mullen 1994).

For the period 1994–2003, waste generator forecasts, discussions with knowledgeable TAN operations personnel, and a draft environmental assessment (DOE 1994b) were the principal sources of information.

Descriptions of Waste Streams. The TAN waste for 1984–1993 was divided into 12 waste streams (see Table 2-2). Some of these streams were extensions of the waste streams identified in the previous report (LITCO 1995) on historical (1952–1983) waste burial.

The TAN waste for 1994–2003 was divided into five waste streams (see Table 2-3). Again, some of these streams are extensions of previous waste streams.

Table 2-2. Waste streams generated at Test Area North during 1984–1993.

Waste stream number	Description of waste
TAN-603-2R	Noncompactible waste resulting from sump cleanout operations
TAN-607-6R	Hot cell/hot shop waste
TAN-623-1R	Sewage sludge
TAN-629-1R	Hangar cleanout to prepare for SMC move-in
TAN-629-2R	Sand-blast grit contaminated with depleted uranium and scrap pieces of depleted uranium from SMC operations
TAN-630-1R	Compactible waste
TAN-647-1R	Unneeded and stored equipment
TAN-650-1R	Compactible waste generated following LOFT tests
TAN-650-2R	LOFT LP-FP-2 resins
TAN-650-3R	LOFT facility parts and systems
TAN-668-1R	Contaminated soil from construction of heavy equipment cleaning facility at entrance to hot shop
TAN-DFN-1R	Waste generated in a decontamination facility

Table 2-3. Waste streams expected to be generated at Test Area North during 1994–2003.

Waste stream number	Description of waste
TAN-607-6P	Hot cell/hot shop waste
TAN-623-1P	Sewage sludge
TAN-629-2P	Sand-blast grit contaminated with depleted uranium and scrap pieces of depleted uranium from SMC operations
TAN-629-3P	Contact-handled waste for incineration
TAN-629-4P	Compactible waste from SMC

2.5.2 Test Reactor Area

The Generator. TRA is located approximately 5 mi north of CFA and approximately 2 mi west of ICPP at the INEL (see Figure 2-1). During the recent time period, the major operating facility at TRA has been the Advanced Test Reactor (ATR), which has been in operation since 1969. In addition to the ATR and its support facilities, the following facilities and laboratories have been or are currently operating at TRA:

- Materials Test Reactor (MTR) (1952–1970)
- Engineering Test Reactor (ETR) (1957–1981)
- TRA hot cells (1952 to present)
- Radiation Measurements Laboratory (RML) (1952 to present)
- Nuclear physics laboratories (1953 to present)
- Radiochemistry laboratories (1952 to present)
- Advanced Test Reactor Critical (ATRC) (1968 to present)
- Engineering Test Reactor Critical (ETRC) (1957–1980)
- Reactivity Measurements Facility (RMF) (1956–1960)
- Advanced Reactivity Measurement Facility (ARMF) (1960–1992)
- Gamma facility
- Metallurgical laboratories
- Hydraulics test facility
- Nuclear materials inspection storage facility
- Maintenance shops.

The ATR uses highly enriched uranium (i.e., 93% U-235 by mass) as its nuclear fuel. The fuel is contained in fuel element assemblies that are composed of multiple fuel plates. The central core of each fuel plate contains a matrix of uranium and aluminum called UAL_x, and is covered by an outer layer of pure aluminum. The reactor core is cooled and neutron-moderated with water. The ATR has a beryllium reflector that surrounds the reactor core. This beryllium is replaced approximately every 6 years; therefore, a large quantity of it has been and will be disposed of at the RWMC.

Each reactor and each critical facility has a canal that is used for storing irradiated and unirradiated fuel and irradiated experiment assemblies. Irradiated fuel is stored in the facility canal for a cooling period and then shipped to ICPP for processing.

The major role of a test reactor is to test the physical, chemical, and nuclear properties of materials during and after exposure to highly intense neutron/gamma fields. Experiments are placed in the reactor core or in the reflector adjacent to the reactor core. The size of the experiments varies from a small irradiation capsule to a major irradiation loop. The standard loop experiment consists of a pressurized water piping system with its own cleanup system, and is designed to provide the controlled physical and chemical conditions for the test region. Typical conditions that are monitored and controlled include the temperature, pressure, and pH of the experiment coolant. The major sponsors of the test reactors have been and continue to be the Bettis Atomic Power Laboratory and the Knolls Atomic Power Laboratory, funded by the Naval Reactors Program (NRP) of DOE and its predecessor agencies. Experiments from these users are designed as specified by the sponsor. After completion of the irradiation, the test internals are generally transferred to the sponsor's facilities for disassembly and examination, or to the TRA hot cells.

The ATR (TRA-670) began full-power operation in 1969. Its core is in the shape of a four-leaf clover. There are nine major regions for experiments. The power for each region can be tailored to meet the experimenter's requirements. The maximum power level of the ATR is 250 MW; however, it typically operates at a power level of about 125 MW. The core loading for the ATR is approximately 40 kg of U-235. During the recent period, it has been necessary to change the beryllium reflector and core internals approximately every 6 years.

From 1969 to 1992, the ATR was operated almost exclusively for the NRP. Since 1992, there has been some diversity in the experiments conducted in the ATR; however, the NRP still remains the primary user of the facility. In addition to NRP experiments, isotope production experiments and experiments for the New Production Reactor Program have been conducted.

To support reactor safety assurance and experiment needs, the ATR has a critical assembly, ATRC, which is a nuclear mockup of the ATR. The major function of the ATRC is to measure reactor criticality and the effect that experiments would have on criticality. The ATRC operates at low power levels (less than 1 kW). At these power levels, the fuel and core structural parts can be handled without the use of remote-handling equipment or shielding.

The RMF and its successor, the ARMF, were designed to be critical assemblies for precisely measuring the neutron cross-section of materials slated for use in or produced by reactors. The RMF was located in the canal of the MTR and used unirradiated MTR fuel elements. It typically operated at less than 100 W. The ARMF replaced the RMF and was located in a separate building (TRA-660) east of the MTR building. The ARMF contains two critical assemblies, ARMF-I and ARMF-II, which share a common canal. In 1969, ARMF-II was reconfigured to support the fast reactor development program. A block of U-238 was placed in the center of the core. After this conversion, the ARMF-II was renamed the Coupled Fast Reactivity Measurement Facility. In 1992, these reactors were placed on temporary inactive status.

The TRA hot cells have been an integral part of test reactor support operations since the early 1960s. They are used for disassembly and examination of irradiated samples and experiment assemblies from the test reactors. During this period of time, almost all naval loop experiment assemblies were and are shipped to the Expended Core Facility (ECF) at NRF for disassembly and examination following irradiation in the ATR. The TRA hot cells were and are used to support the test reactor non-naval loop experiments. One major support operation was and is the processing of cobalt and iridium produced in the ATR in their isotope production program.

Future operations at TRA are expected to be somewhat different from those of today. The naval loop experiments and isotope production will continue; however, the mix of operations is expected to change.

Generation of the Waste. The waste shipped from TRA to the RWMC comes primarily from the operation of the ATR and the examination of irradiated experiment assemblies in the TRA hot cells. This radioactive waste contains radioactive fission products produced in the nuclear fuel and radionuclides produced by neutron activation. The nuclear fuel-produced radioactivity is typically classified as MFP; however, some activation products are associated with certain fuels. Neutron activation products are typically classified as MAP. The actual distribution of specific nuclides in either MFP or MAP depends on the reactor fuel and the process that generated the waste.

Until 1992, the irradiated fuel was reprocessed at ICPP; since then, it has been placed into storage at ICPP. The irradiated fuel is normally sent as intact assemblies to ICPP. Therefore, only a minor component of the activity produced at the ATR is left at TRA. This component is the result of fission products leaking through the reactor fuel cladding into the reactor coolant. Once in the coolant, the fission products can potentially contaminate all items in contact with the coolant. This includes materials inside the reactor vessel, and pipes, pumps, and cleanup systems associated with the primary coolant.

In addition to fuel leakage, there can also be leakage of radioactivity from the fueled experiments. This primarily contaminates the experiment coolant and cleanup system, and secondarily contaminates the main reactor coolant. When these experiments are disassembled in the TRA hot cells, the irradiated components and the handling equipment, along with rags, etc., are contaminated and become waste.

The filters in the reactor and hot cell ventilation systems also contain some of the fission products produced in the reactor fuel and fueled experiments. Although the reactors are water-cooled, there have been experiments in which the coolant has been gaseous. In those cases, the filters from cleanup systems of those experiments were contaminated and eventually sent to the RWMC.

Activation products are produced when neutrons are captured or otherwise interact to produce radionuclides. Neutron interactions can occur in the reactor fuel, and the radionuclides are carried along with the fission products. Neutrons can also interact with reactor and experiment structural components, resulting in radionuclides becoming fixed contamination in those components and also through corrosion in the reactor or experiment coolant. Once in the coolant system, the radionuclides can potentially contaminate the same items as the fission

products. Therefore, for radioactive waste generated by test reactor operations and support activities, there will be a mixture of fission products and activation products.

In addition to fission products and activation products, TRU radionuclides are produced in a reactor. These radionuclides are produced by multiple neutron capture events, combined with beta and alpha decay. The test reactors were used to produce minor amounts of TRU radionuclides, generally in the microcurie range. Most TRU radionuclides not bound in reactor fuel were brought here from offsite producers.

The hot cells are the second largest generator of waste at TRA. In addition to experiments in the test reactors, the TRA hot cells have been used to process experiments performed outside TRA. These include the severe fuel damage experiments performed at the Power Burst Facility (PBF) and fuel samples from the damaged TMI-2 reactor. The PBF and TMI fuel contains low-enriched uranium (approximately 4% U-235 by mass). The radionuclide distributions in these fuels are different from those in the test reactor fuels. In addition, the activation products due to the zircaloy cladding are different from activation products generated by test reactors.

Almost all items removed from the hot cells are considered to be radioactive. If there is no further need for these items, they are classified as radioactive waste.

The critical facilities (i.e., ATRC and ARMF) contribute small amounts of radioactive waste, most of which is carried into the facilities on samples and experiments from the test reactors or from non-TRA facilities. In most cases, the mix between fission products and activation products is about the same as that found for the test reactors.

The radiochemistry and physics laboratories at TRA handle small quantities of radioactive materials as part of their research, typically on the order of microcuries to millicuries. The hot cell and the californium cell in the alpha wing (TRA-661) are possible exceptions. The radiochemistry programs have used the alpha-wing hot cell for the separation of transuranics and other research efforts. The californium cell contains microgram amounts of Cf-252 used to produce nanocurie amounts of fission products for nuclear decay measurements. The alpha-wing solid waste has higher concentrations of alpha-emitters from the decay of TRU nuclides. The remainder of the radiochemistry and physics laboratories generate waste similar in content to reactor plant waste.

The gamma facility operating during this time period uses irradiated fuel elements from the ATR and is located in the ATR canal. Waste from this gamma facility is included with the ATR canal waste.

Radioactive liquid waste from TRA was disposed of in the TRA waste retention basins (if low to moderate activity) or sent to ICPP for processing (if moderate to high activity).

With respect to nonradiological contaminants in TRA waste sent to the RWMC, the major generator was the ATR. The primary contaminant is beryllium. This waste is generated when a reactor reflector is replaced.

The following are examples of waste streams sent to the RWMC from TRA:

- Ion-exchange resins used in the reactor coolant systems
- Core and experiment loop components constructed of aluminum or stainless steel; they generally contain activation products
- Contaminated rags and floor sweepings
- Contaminated concrete, bricks, and wood
- Irradiated beryllium from the reactor reflector changeouts
- Contaminated sludge from the resin beds, etc.

Waste to be generated from 1994–2003 is expected to be similar to that generated from 1984–1993.

General Availability of Information. The main sources of data pertaining to waste shipments from TRA in the period 1984–1993 are RWMIS and selected technical reports and letters. The RWMIS values and those in annual waste management reports for 1984–1993 agree reasonably well.

Projections for waste expected to be shipped from TRA in the period 1994–2003 are available in a letter (Keating 1993). The letter forecasts waste volumes, total activities, and radionuclide distributions for various waste streams.

Data-Collection Approach. The approach selected for data collection for TRA waste for the period 1983–1994 was to use RWMIS entries and shipping records for the waste volumes and radioactivity. These sources were then supplemented with other information sources as necessary to develop or refine the radionuclide distributions. The other sources were various reports referenced herein, as well as nuclear physics considerations and calculations.

Reports and shipping records provide varying degrees of completeness in specifying radionuclide distributions. The following information describes how the available records and reports were combined with nuclear physics evaluations to project a reasonably complete distribution of radionuclides having the appropriate total amount of radioactivity.

Table 2-4 is the master list of radionuclides considered in estimating the nuclide-by-nuclide activity breakdown of the waste generated at TRA. This list is a composite based on (a) a performance assessment of dose at the RWMC performed in 1993, (b) the reporting requirements imposed by the NRC on waste from operating commercial power reactors (10 CFR 61), and (c) their expected importance in TRA waste. Based on an activity buildup calculation using the ORIGEN2 computer code (Graff 1980; Schnitzler 1994) for a typical ATR fuel element irradiation history, the activity for any radionuclide with an atomic number or mass greater than that of Cm-244 is too weak to be reportable and is not included.

Table 2-4. Master list of radionuclides evaluated for waste from the Test Reactor Area.

Nuclide	Half-life[a] (years)	Decay mode	Fission product	Activation product
Am-241	433	α	–	X
C-14	5.7×10^3	β	–	X
Ce-144	0.78	$\beta+\gamma$	X	–
Co-60	5.3	$\beta+\gamma$	–	X
Cm-242	0.45	α	–	X
Cm-244	18.1	α	–	X
Cs-137	30.2	$\beta+\gamma$	X	–
Fe-55	2.73	β	–	X
Eu-152	13.5	$\beta+\gamma$	X	–
Eu-154	8.6	$\beta+\gamma$	X	–
Eu-155	4.7	$\beta+\gamma$	X	–
H-3	12.3	β	X	X
I-129	1.6×10^7	$\beta+\gamma$	X	–
Nb-94	2.0×10^4	β	–	X
Ni-59	7.6×10^4	β	–	X
Ni-63	100	β	–	X
Np-237	2.1×10^6	α	–	X
Pu-238	87.7	α	–	X
Pu-239	2.4×10^4	α	–	X
Pu-240	6.6×10^3	α, sf	–	X

Table 2-4. (continued).

Nuclide	Half-life[a] (years)	Decay mode	Fission product	Activation product
Pu-241	14.4	β	—	X
Ra-226	1.6×10^3	α	—	—
Sb-125	2.8	β+γ	X	X
Sr-90	29	β	X	—
Tc-99	2.1×10^5	β	X	—
U-232	70	α	—	—
U-233	1.6×10^5	α	—	—
U-234	2.5×10^5	α	—	—
U-235	7.0×10^8	α	—	—
U-236	2.3×10^7	α	—	—
U-238	4.5×10^9	α	—	—

a. Half-lives taken from General Electric Company (1989).

α — Decays by alpha emission
β — Decays by beta emission
β+γ — Decays by beta emission plus gamma transitions
α, sf — Decays by alpha emission and spontaneous fission

Radioactive waste generated at TRA has been reported as individual nuclides, MFP, MAP, unidentified beta-gamma, or unidentified alpha. Most waste streams or waste generation processes at TRA contain all types of activity; however, the relative mix differs. Because there are different mixes, it was decided that the waste should be categorized according to the generator mode or generic content, rather than by activity. Based on a review of commercial power plant waste reports (e.g., EPRI 1987), three general categories of waste were identified by analogy for TRA:

- Dry radioactive waste not otherwise specified
- Sludge
- Reactor coolant resin.

The radionuclides for each of the waste categories and the respective activity scaling factors (radioactivity distributions; see Section 5.4.3 for a more detailed discussion) for each waste category are listed in Tables 2-5 through 2-7. INEL data for the scaling factors of difficult-to-measure radionuclides in TRA waste are limited. Therefore, many of the scaling factors for these radionuclides were taken from data gathered (EPRI 1987) on commercial nuclear power reactors. There are limitations in applying those data to waste from INEL test reactors, but these data are the most applicable data that are available.

The scaling factors are based on fractional activities consistent with the assumption that measurement of total activity via the G-M method would include only gamma activity. (The G-M method and its limitations are discussed in detail in Section 5.) The approach followed to generate tables for which more than one data source was used is described in Harker and Akers (1994) and in Harker (1995a).

For the years 1984–1993, annual reports, topical reports, and letters that indicate waste-generating activities, as well as RWMIS, were used. The following approach was used:

- *Reports containing nuclide-by-nuclide distributions.* The individual activities, as they are listed in the report, were used. In most cases, if an error was not stated, an assumed measurement error was assigned.
- *Reports containing nuclide-by-nuclide distributions plus MAP, MFP, unidentified beta-gamma, and/or unidentified alpha.* The waste was identified as one of the three waste categories listed above. The MAP, MFP, and beta-gamma activities were summed to get a total activity of overall gamma-emitters. A nuclide-by-nuclide distribution was calculated based on this total gamma activity and the corresponding activity scaling factors for that waste category. The explicit nuclide distribution, the calculated waste category nuclide distribution, and the calculated alpha nuclide distribution were all added as separate tables.

Table 2-5. Nuclides and activity scaling factors for dry radioactive waste from the Test Reactor Area.

Nuclide	Activity[a] scaling factor	Data source
H-3[b]	8.2×10^{-2}	EPRI (1987)[c]
C-14	1.1×10^{-3}	EPRI (1987)[c]
Fe-55	1.9	EPRI (1987)[c]
Co-60	6.7×10^{-1}	EPRI (1987)[c]
Ni-59	5.7×10^{-4}	Evans et al. (1984)[d]
Ni-63	3.2×10^{-1}	EPRI (1987)[c]
Sr-90	9.2×10^{-4}	EPRI (1987)[c]
Tc-99	1.8×10^{-4}	EPRI (1987)[c]
I-129	4.4×10^{-8}	Harker (1995b)[e]
Cs-137	2.0×10^{-1}	EPRI (1987)[c]
Ce-144	4.7×10^{-3}	EPRI (1987)[c]
Eu-154	2.9×10^{-6}	Evans et al. (1984)[d]
Eu-155	9.4×10^{-3}	Graff (1980) and Schnitzler (1994)[f]
U-234	2.1×10^{-6}	Graff (1980) and Schnitzler (1994)[f]
U-235	4.5×10^{-8}	Graff (1980) and Schnitzler (1994)[f]
U-236	8.0×10^{-7}	Graff (1980) and Schnitzler (1994)[f]
Np-237	1.2×10^{-6}	Graff (1980) and Schnitzler (1994)[f]
Pu-238	5.4×10^{-5}	EPRI (1987)[c]
Pu-239	5.4×10^{-5}	EPRI (1987)[c]
Pu-240	5.6×10^{-6}	Graff (1980) and Schnitzler (1994)[f]
Pu-241	5.9×10^{-3}	EPRI (1987)[c]

Table 2-5. (continued).

Nuclide	Activity[a] scaling factor	Data source
Am-241	2.7×10^{-5}	EPRI (1987)[c]
Cm-242	2.7×10^{-5}	EPRI (1987)[c]
Cm-244	2.5×10^{-5}	EPRI (1987)[c]

a. It was assumed that the measured activity was determined by a predominantly gamma-sensitive device (e.g., G-M counter, NaI scintillation detector). As such, only gamma activity was reported. However, there were cases where the gamma activity was increased to account for the beta-emitters. This correction was not universally applied and, in many cases, not even noted. To be conservative, it was assumed that the reported activity (MAP, MFP, unidentified beta-gama) includes only gamma activity. The scaling factors listed in this table have taken the beta activity into account. Therefore, the sum of the scaling factors is a number greater than unity. The difference between the sum and unity is the relative beta activity. See Harker (1995a) for details on how the scaling factors were derived.

b. There is a question as to whether tritium is present at the fraction indicated in this table for all dry waste. It is present in dry waste that has direct contact with the reactor coolant and, in some cases, where there has been secondary contact. The tritium scaling factor listed in this table represents a history of experience with pressurized water reactors and should give numbers that are valid on the average. However, in those cases where there was evidence that tritium was not present or was present in much lower concentrations, the scaling factor for tritium in the table was not used. A note to this effect was placed with that data entry.

c. Dry active waste generated by all commercial pressurized water reactors in the United States.

d. Activation products in 304 stainless steel.

e. The factor was based on the ratio of I-131 activity to Cs-137 activity in ATR primary coolant water and on ratios of I-129, I-131, and Cs-137 activities calculated with ORIGEN2.

f. ORIGEN2 calculation based on irradiating an ATR fuel element for 85 days at 8 MW per element and allowing it to decay for 1 year.

Table 2-6. Nuclides and activity scaling factors for sludge waste from the Test Reactor Area.

Nuclide	Activity[a] scaling factor	Data source
H-3[b]	1.0×10^{-1}	EPRI (1987)[c]
C-14	1.0×10^{-2}	EPRI (1987)[c]
Fe-55	7.7×10^{-1}	EPRI (1987)[c]
Co-60	8.7×10^{-1}	EPRI (1987)[c]
Ni-59	4.0×10^{-3}	Resin[d]
Ni-63	3.2×10^{-1}	EPRI (1987)[c]
Sr-90	6.4×10^{-4}	EPRI (1987)[c]
Tc-99	1.1×10^{-4}	EPRI (1987)[c]
I-129	3.0×10^{-8}	Harker (1995b)[e]
Cs-137	1.3×10^{-1}	EPRI (1987)[c]
Ce-144	4.2×10^{-2}	EPRI (1987)[c]
Eu-154	3.1×10^{-3}	Resin[d]
Eu-155	1.3×10^{-3}	Resin[d]
U-234	2.2×10^{-6}	Resin[d]
U-235	4.8×10^{-8}	Resin[d]
U-236	8.5×10^{-7}	Resin[d]
Np-237	1.3×10^{-6}	Resin[d]
Pu-238	3.3×10^{-5}	EPRI (1987)[c]
Pu-239	3.4×10^{-5}	EPRI (1987)[c]
Pu-240	2.1×10^{-5}	Resin[d]
Pu-241	3.7×10^{-3}	EPRI (1987)[c]

Table 2-6. (continued).

Nuclide	Activity[a] scaling factor	Data source
Am-241	1.4×10^{-5}	EPRI (1987)[c]
Cm-242	2.7×10^{-5}	EPRI (1987)[c]
Cm-244	1.3×10^{-5}	EPRI (1987)[c]
U-238	2.9×10^{-9}	Graff (1980) and Schnitzler (1994)[f]

a. It was assumed that the measured activity was determined by a predominantly gamma-sensitive device (e.g., G-M counter, NaI scintillation detector). As such, only gamma activity was reported. However, there were cases where the gamma activity was increased to account for the beta-emitters. This correction was not universally applied and, in many cases, not even noted. To be conservative, it was assumed that the reported activity (MAP, MFP, unidentified beta-gama) includes only gamma activity. The scaling factors listed in this table have taken the beta activity into account. Therefore, the sum of the scaling factors is a number greater than unity. The difference between the sum and unity is the relative beta activity. See Harker (1995a) for details on how the scaling factors were derived.

b. There is a question as to whether tritium is present at the fraction indicated in this table for all dry waste. It is present in dry waste that has direct contact with the reactor coolant and, in some cases, where there has been secondary contact. The tritium scaling factor listed in this table represents a history of experience with pressurized water reactors and should give numbers that are valid on the average. However, in those cases where there was evidence that tritium was not present or was present in much lower concentrations, the scaling factor for tritium in the table was not used. A note to this effect was placed with that data entry.

c. Sludge waste generated by all commercial pressurized water reactors in the United States.

d. Activities relative to Co-60 (for Ni-59), to Cs-137 (for Eu-154, Eu-155, U-234, U-235, U-236, Np-237), or to Pu-239 (for Pu-240) are assumed to be the same as those listed for resins (see Table 2-7).

e. The factor was based on the ratio of I-131 activity to Cs-137 activity in ATR primary coolant water and on ratios of I-129, I-131, and Cs-137 activities calculated with ORIGEN2.

f. ORIGEN2 calculation based on an ATR fuel element irradiated for 85 days at 8 MW per element and allowed to decay for 1 year. Activity is scaled to Pu-239 activity as measured by gamma spectrometry.

Table 2-7. Nuclides and activity scaling factors for reactor coolant resin from the Test Reactor Area.

Nuclide	Activity[a] scaling factor	Data source
H-3	5.0×10^{-4}	ATR[b]
C-14	4.3×10^{-3}	EPRI (1987)[c]
Fe-55	2.0×10^{-1}	EPRI (1987)
Ni-59	2.8×10^{-3}	ATR[b]
Ni-63	2.8×10^{-1}	ATR[b]
Co-60	6.8×10^{-1}	ATR[b]
Sr-90	2.8×10^{-1}	ATR[b]
Tc-99	1.5×10^{-5}	ATR[b]
I-129	6.8×10^{-8}	Harker (1995b)[d]
Cs-137	3.1×10^{-1}	ATR[b]
Ce-144	6.7×10^{-3}	ATR[b]
Eu-154	7.3×10^{-3}	ATR[b]
Eu-155	3.1×10^{-3}	ATR[b]
U-234	4.2×10^{-6}	Graff (1980) and Schnitzler (1994)[e]
U-235	9.2×10^{-8}	Graff (1980) and Schnitzler (1994)[e]
U-236	1.6×10^{-6}	Graff (1980) and Schnitzler (1994)[e]
Np-237	2.6×10^{-6}	Graff (1980) and Schnitzler (1994)[e]
Pu-238	1.8×10^{-4}	ATR[b]
Pu-239	4.6×10^{-5}	ATR[b]
Pu-240	2.8×10^{-5}	Graff (1980) and Schnitzler (1994)[e]
Pu-241	1.5×10^{-2}	ATR[b]

Table 2-7. (continued).

Nuclide	Activity[a] scaling factor	Data source
Am-241	4.2×10^{-3}	ATR[b]
Cm-242	2.8×10^{-4}	ATR[b]
Cm-244	1.3×10^{-4}	ATR[b]

a. It was assumed that the measured activity was determined by a predominantly gamma-sensitive device (e.g., G-M counter, NaI scintillation detector). As such, only gamma activity was reported. However, there were cases where the gamma activity was increased to account for the beta-emitters. This correction was not universally applied, and in many cases, not even noted. To be conservative, it was assumed that the reported activity (MAP, MFP, unidentified beta-gama) includes only gamma activity. The scaling factors listed in this table have taken the beta activity into account. Therefore, the sum of the scaling factors is a number greater than unity. The difference between the sum and unity is the relative beta activity. See Harker (1995a) for details on how the scaling factors were derived.

b. Activities scaled to Cs-137 (for H-3, Sr-90, Tc-99, Eu-154, Eu-155), to Co-60 (for Ni-59, Ni-63), or to Pu-239 (for Pu-238, Pu-241, Am-241, Cm-242, Cm-244) as measured for ATR resin shipment 92026 (see Harker and Akers 1994) are assumed to be representative for all resin shipments.

c. Assumed reactor coolant resin C-14 activity relative to Co-60 as reported for pressurized water reactors is representative of the ATR resin C-14 to Co-60 activity ratio.

d. The factor was based on the ratio of I-131 activity to Cs-137 activity in ATR primary coolant water and on ratios of I-129, I-131, and Cs-137 activities calculated with ORIGEN2.

e. ORIGEN2 calculation based on an ATR fuel element irradiated for 85 days at 8 MW per element and allowed to decay for 1 year. Activity is scaled to Pu-239 activity as measured by gamma spectrometry.

As the preceding and following discussions imply, radionuclide distributions were developed from process knowledge and nuclear physics calculations for each category of waste stream. Therefore, no single, uniform assumption was used for the distribution of generic radioactivity terms such as MAP and MFP in shipping records.

It is noted that, for some TRA waste streams, two entries for the same radionuclide in the same year appear on the datasheets. In such cases, the bounds are different on the different entries because parts of the total activity were determined by different methods. For example, one entry for Cs-137 in a given stream for a given year was obtained from laboratory measurements. Another entry for Cs-137 for the same stream and year was obtained by distributing a MFP term using scaling factors.

The approach selected for data collection for TRA waste for the period 1994–2003 was to use one of two methods, depending on the waste stream. Either (a) the activity values in the generator forecast (Keating 1993) were used, or (b) the RWMIS values for the recent period (1984–1993) were used to obtain the average annual activity and this value was assumed to apply also for the projected waste. The *INEL D&D Long-Range Plan* (Buckland et al. 1993) was also used as an information source for waste to be generated by D&D activities at TRA.

Description of Waste Streams. The TRA waste for 1984–1993 was divided into 14 waste streams (see Table 2-8). Many of these streams were extensions of some of the waste streams identified in the previous report on historical (1952–1983) waste burial.

The TRA waste for 1994–2003 was divided into eight waste streams (see Table 2-9), including three streams of D&D waste from TRA. Again, many of these streams are extensions of previous waste streams.

2.5.3 Idaho Chemical Processing Plant

The Generator. ICPP is located near the center of the INEL between CFA and TRA (see Figure 2-1). The primary purpose of this facility was to recover U-235 from expended military and test reactor fuels.

The facility originally included a process building containing dissolvers to dissolve the fuel assemblies in nitric and hydrofluoric acids, and a solvent extraction system that used tributyl phosphate, hexone, and nitric acid to recover the uranium. Laboratory, water treatment, and evaporator facilities were also part of the complex. A storage pool, housed in a separate building, to store the fuel under water until a processing campaign was underway was also part of the system.

In the early 1960s, a fluidized bed waste calciner was constructed and operated to convert the highly radioactive liquid waste, resulting from processing the fuel, to a dry, granular, solid. This was replaced by the New Waste Calcining Facility (NWCF) in the late 1970s.

Also, in the early 1970s, an improved and larger fuel storage pool was constructed and placed into operation. About 1973, a new building was joined onto the original fuel storage facility to store dry graphite-type fuel, for which no uranium recovery process existed.

Table 2-8. Waste streams generated at the Test Reactor Area during 1984–1993.

Waste stream number	Description of waste
TRA-603-1R	Resins
TRA-603-4R	Core and loop components
TRA-603-6R	Sludge
TRA-603-7R	Glass
TRA-603-15R	Metal (aluminum, stainless steel, etc.)
TRA-603-18R	Rags, floor sweepings, and glassware
TRA-603-21R	Construction materials, concrete, brick, sand, soil, and asphalt
TRA-603-26R	Lead
TRA-603-27R	Noncompactible waste such as glass and metal
TRA-603-28R	Combination of glass, halogenated plastic, absorbed liquid, metal chips, wire, etc.
TRA-632-2R	Hot cell waste consisting of small amounts of metal, glassware, plastic bottles, etc.
TRA-632-3R	Absorbed liquids
TRA-642-7R	Various combustible materials such as paper, wood, plastic, rags, and wipes
TRA-670-1R	Beryllium reflectors from the ATR

Table 2-9. Waste streams expected to be generated at the Test Reactor Area during 1994–2003.

Waste stream number	Description of waste
TRA-603-1P	Resins
TRA-603-4P	Core and loop components
TRA-603-15P	Metal (aluminum, stainless steel, etc.)
TRA-603-27P	Noncompactible waste such as glass and metal
TRA-670-1P	Beryllium reflectors from the ATR
D+DTRA-603-1P	Construction material such as soil, concrete, and rubble
D+DTRA-603-2P	Metal (aluminum, stainless steel, etc.)
D+DTRA-603-3P	Noncompactible waste such as glass and metal

During the period 1984–1993, expended fuels were received, stored, and processed, and the liquid waste was calcined. The dry, granular solids produced by calcination were stored in underground, stainless-steel bins, where they will remain until a process is developed to convert these solids to a nonleachable form, such as a glass, and the resulting product is placed in a permanent repository.

In 1992, a decision was made by DOE to discontinue processing of all fuels at ICPP. Currently and in the projected period (1994–2003), the principal operations at ICPP are limited to the storage of fuel and the calcination of high-level liquid waste.

A project is underway to move all of the fuel stored in the old fuel storage basin to the newer fuel storage basin. Also, efforts are underway to develop a flowsheet to calcine the stored, sodium-containing, liquid waste, which cannot be calcined using the flowsheet developed for the aluminum- and zirconium-containing waste calcined in the past. If calcination cannot be accomplished, it will be necessary to develop an alternative method to solidify the sodium-containing waste. New projects are being considered for implementation as the mission of ICPP changes.

Generation of the Waste. Most of the radioactive waste produced at ICPP remains in storage at that facility. Raffinates resulting from the dissolution and recovery of uranium from the expended nuclear fuels, waste solutions resulting from the decontamination of process cells and equipment, and waste solutions produced by concentrating radioactive liquids in the process equipment waste evaporator are stored in underground stainless-steel tanks. This waste is later processed in the fluidized bed calciner at ICPP to convert the liquid to a granular solid. The solids are stored in underground, stainless-steel bins.

The processing of irradiated fuels produced thousands of gallons of high-level radioactive liquid waste containing several million curies of radionuclides. These radionuclides are nearly all retained at ICPP, either as liquid waste stored in the underground stainless-steel tanks or as granular solids stored in the underground stainless-steel bins, where they are being depleted slowly by radioactive decay.

LLW is generated routinely from cleanup operations, equipment modifications and replacement, and decontamination of equipment. This waste includes blotting paper, rags, anti-contamination clothing, plastic bags, glassware, rubber gloves, metal structural materials (usually stainless steel), filters, roofing materials, asphalt, concrete, wood, and occasionally metal vessels, and casks and fuel storage racks that have been decontaminated to a low level of radioactivity. No fuel end boxes, which contributed the majority of radioactivity sent from ICPP to the RWMC at an earlier date, were disposed of during this time period (1984–1993).

Although changes in the mission of ICPP may produce different kinds of high-level radioactive waste in the period 1994–2003, this waste very likely will be stored at ICPP for some time. The waste that will be shipped from ICPP to the RWMC will probably not be substantially different from that produced in the 1984–1993 period. It will probably consist of cleanup materials such as rags and blotting paper, and construction materials such as steel and concrete from decontamination and decommissioning operations and modifications to existing structures.

General Availability of Information. The primary source of information available for the ICPP waste is the RWMIS printouts for 1984–1993, and technical note printouts compiled from the RWMIS records for the same time period. These technical note printouts summarized the RWMIS values for all of the waste shipments from each building. They provided the total curies, radionuclide distribution, type of container, volume, and weight. A content code that described the type of material in each waste shipment was also available from RWMIS records for each building.

Projections for waste expected to be shipped from ICPP in the period 1994–2003 are available primarily from ICPP solid radioactive waste forecasts. These forecasts are essentially extensions for waste resulting from existing or planned operations, and could change substantially as the mission of ICPP changes.

Data-Collection Approach. The approach to collect data for the period 1984–1993 was to obtain RWMIS technical note printouts for each building at ICPP, and to record the volume, total curies, amounts of various radionuclides, and type of waste container used for each building. Then, the content codes for the waste from each building were obtained from the RWMIS records, and the description of the waste was obtained by using these content codes. Contacts with knowledgeable individuals were made to provide additional detail about the waste.

Data for the projected waste (1994–2003) were obtained from the solid radioactive waste forecasts for ICPP. Curie values for the individual radionuclides were obtained from these forecasts. The waste descriptions were obtained either from the forecasts or from individuals who prepared the forecasts.

Description of Waste Streams. The ICPP waste for the period 1984–1993 was divided into eight waste streams (see Table 2-10).

One waste stream is expected to be produced at ICPP for the period 1994–2003 (see Table 2-11).

2.5.4 Naval Reactors Facility

The Generator. NRF is located in the western part of the INEL about 14 mi north-northeast of the RWMC (see Figure 2-1).

NRF was established in 1950 when construction began on the prototype power plant for the U.S. Navy's first nuclear-powered submarine, the USS NAUTILUS. This prototype, later named S1W, was developed to test the propulsion plant design and to train Navy personnel to operate reactors in preparation for duty on nuclear-powered submarines and ships in the fleet. Two additional naval reactors prototypes were subsequently built at NRF, A1W in 1957 and S5G in 1965. The basic mission of these other prototypes was the same as for the original prototype—to test propulsion plant designs and to train Navy personnel. The S1W plant was shut down in October 1989, the A1W plant was shut down in January 1994, and the S5G plant was shut down in May 1995.

ECF, built at NRF in 1958, was designed to receive irradiated naval reactor fuel, perform examinations on the fuel elements, remove excess structural material from the fuel elements, and transfer the fuel elements to ICPP. ECF has also received and examined naval fuel test specimens that have been irradiated in other reactors, such as the ATR. The fuels are handled remotely under water in the ECF water pits. The water serves as a transparent shielding medium in which a number of procedures can be carried out, including disassembling, cutting, sawing, milling, and visually examining various parts of the fuel elements. Some procedures are also carried out in hot cells at ECF.

Future operations at NRF will change somewhat from past operations. There will be no operating reactor plants at NRF. A1W and S5G will be undergoing defueling and inactivation operations. ECF will likely continue to examine spent fuel elements from naval cores.

Generation of the Waste. LLW is generated by the naval reactor prototypes as a result of activities such as reactor coolant sampling, maintenance, repair, inactivation, and refueling/defueling actions that require interface with the contaminated plant internals. LLW is generated at ECF as a result of fuel examination work. The majority of the waste originating at the prototype plants is compactible and largely incinerable waste (e.g., plastic bags, rubber gloves, blotter paper, and other materials used to contain contamination) with very low levels of radioactivity. In addition to this compactible and largely incinerable waste, there have also been occasional metal valves and piping sections that are noncompactible and that can contain higher quantities of radioactivity. Metal tanks and drums containing spent ion-exchange resins and sludge from water reprocessing systems add to the noncompactible component of the waste streams.

Table 2-10. Waste streams generated at the Idaho Chemical Processing Plant during 1984–1993.

Waste stream number	Description of waste
CPP-601-8R	Waste from the fuel processing building resulting from replacement and/or removal of equipment, asbestos, and materials used for decontamination of the facility
CPP-604-2R	Pieces of equipment, insulated piping, contaminated soil, asbestos, and cleaning materials such as cloth, paper, plastic, etc.
CPP-605-1R	High-efficiency particulate air (HEPA) filters, metal, soil, concrete, cleanup materials such as paper, cloth, plastic, etc., from the facility that contains the off-gas cleanup system
CPP-610-1R	Building materials, equipment, cleanup materials, metal, filters, fuel storage racks, and lead sheets and bricks, all from various areas
CPP-649-1R	Filters from the Atmospheric Protection System, and construction materials from upgrading of the facility
CPP-659-1R	Metals, construction materials, and cleanup materials such as paper, cloth, plastic, and concrete from the NWCF
CPP-666-1R	Waste from 16 buildings, mostly construction materials, fuel charging casks, cleanup materials, and liquids solidified in Portland cement
CPP-684-1R	Waste from the Remote Analytical Facility: construction materials and cleanup materials

Table 2-11. Waste streams expected to be generated at the Idaho Chemical Processing Plant during 1994–2003.

Waste stream number	Description of waste
CPP-All-1P	Contaminated structural materials, mostly metals and concrete, and cleanup materials, such as blotting paper, rags, anti-contamination clothing, plastic, etc. Waste from all areas at ICPP will be shipped directly to the RWMC. Other combustible and noncombustible waste will be shipped to the Waste Experimental Reduction Facility (WERF) for processing and reported as WERF waste.

The majority of both the radioactivity and the volume of waste that is transferred from NRF to the RWMC comes from ECF. Most of the radioactivity emerging from ECF is in highly corrosion-resistant metal structural materials removed during the naval fuel examinations. This material is loaded into metal containers which, in turn, fit into large shielded shipping casks. These casks are then taken to the RWMC, where the metal containers are removed and buried.

The only difference in the NRF waste between the time periods 1984–1993 and 1994–2003 is in the quantities of waste shipped. The operations and processes that give rise to these waste streams are substantially the same as in the past.

General Availability of Information. The main sources of data pertaining to waste shipments from NRF in the period 1984–1993 are Bartolomucci (1989) and Nieslanik (1994). The Nieslanik letter, in turn, relies on the RWMIS database of shipping and disposal records, an internal NRF database, shipping records, technical work record books, and shipping logs. These two letters document efforts made by NRF to improve the information available on (a) the distribution of radionuclides within the identified NRF waste streams and (b) the total radioactivity shipped from NRF in the scrap casks during 1984–1993.

The method used by NRF to determine the total activity and radionuclide distribution in scrap cask inserts shipped from NRF from 1976 through 1989 was outlined in Bartolomucci (1989). This method was based on knowledge of the metal alloys in the reactor core structural materials and the reactor core radiation history. This information allowed NRF to calculate the extent of expected neutron activation of the core structural material. As pointed out in the letter, this technique is similar to the calculation methods used to determine power levels and lifetimes for nuclear cores and has been validated empirically. The same method was used for the scrap cask shipments between 1990 through 1993 that were dealt with in Nieslanik (1994).

Projections for waste expected to be shipped from NRF in the period 1994–2003 are available in a waste generator forecast letter (Charter 1993) that forecasts volumes, activities, and radionuclide distributions for a number of waste streams.

Data-Collection Approach. The approach selected for data collection for NRF waste during the period 1984–1993 was to take data from Bartolomucci (1989), Nieslanik (1994), and the RWMIS database. The figures from all sources were checked against each other. They agreed closely since they all were derived from the same basic set of NRF records.

The approach selected for data collection for the period 1994–2003 was to use the NRF forecasts (Charter 1993), supplemented by information gathered for the 1984–1993 period for those streams expected to continue to be generated.

Description of Waste Streams. The NRF waste for 1984–1993 was divided into six waste streams (see Table 2-12). These streams were extensions of some of the waste streams identified in the previous report on historical (1952–1983) waste burial.

Table 2-12. Waste streams generated at the Naval Reactors Facility during 1984–1993.

Waste stream number	Description of waste
NRF-617-1R	Low-level compactible and noncompactible waste resulting from operation of the prototype reactors and related activities
NRF-617-2R	Lead and asbestos
NRF-618-4R	Structural components removed from Navy nuclear fuel modules (end boxes, etc.), 1984–1988
NRF-618-6R	Solidified sludge, resin, waste liquids in vermiculite, Radioactive Waste Disposal System (RWDS) modules
NRF-618-7R	Low-level compactible and noncompactible waste resulting from work at the ECF water pits and hot cells
NRF-618-8R	Structural components removed from Navy nuclear fuel modules (end boxes, etc.), 1989–1993

The NRF waste for 1994–2003 was divided into three waste streams, again extensions of previous waste streams (see Table 2-13).

As stated previously, the majority of the radioactivity from NRF came from ECF in scrap cask inserts. Three NRF waste streams, NRF-618-4R (1984–1988), NRF-618-8R (1989–1993), and NRF-618-8P (1994–2003), encompass this waste. The waste was divided into these three streams based on the three indicated sequential time periods and because of changes in radionuclide distribution in those time periods.

2.5.5 Argonne National Laboratory—West

The Generator. ANL-W is located in the southeastern part of the INEL, approximately 35 mi west of Idaho Falls (see Figure 2-1).

The mission of ANL-W has been, since the beginning of operation, the research and development of liquid metal cooled reactors and advanced nuclear power plant technology. The primary focus for ANL-W research until 1994 was the Integral Fast Reactor (IFR) Project integrated with an onsite fuel recycling process called pyroprocessing. The objectives were to increase reactor safety, reduce radioactive waste components and concentrations, and improve reactor fuel efficiency.

Table 2-13. Waste streams expected to be generated at the Naval Reactors Facility during 1994–2003.

Waste stream number	Description of waste
NRF-618-6P	Solidified sludge, resin, waste liquids in vermiculite, RWDS modules
NRF-618-7P	Low-level compactible and noncompactible waste resulting from work at the S5G prototype and at the ECF water pits and hot cells
NRF-618-8P	Structural components removed from Navy nuclear fuel modules (end boxes, etc.), 1994–2003

ANL-W consists of seven major complexes: the Experimental Breeder Reactor-II (EBR-II), the Transient Reactor Test Facility (TREAT), the Zero Power Physics Reactor (ZPPR), the Hot Fuel Examination Facility (HFEF), the Fuel Cycle Facility (FCF), the Fuel Manufacturing Facility (FMF), the Laboratory and Office Building (L&O), and support facilities such as the Radioactive Liquid Waste Treatment Facility (RLWTF), the Sodium Components Maintenance Shop (SCMS), and the Sodium Process Facility (SPF).

EBR-II consists of a sodium-cooled reactor with a thermal power rating of 62.5 MW, an intermediate closed loop of secondary sodium, and a steam plant that produces 19 MW of electrical power through a conventional turbine generator. The original emphasis in the design and operation of EBR-II was to demonstrate a complete breeder reactor power plant with onsite reprocessing of metallic fuel. The demonstration was successfully carried out from 1964 to 1969. The emphasis at EBR-II was then shifted to irradiation testing of fuels and materials for future, larger liquid metal reactors. The EBR-II has also been used to provide electrical power for ANL-W and the INEL. The EBR-II cooling tower, SCMS, and SPF are also associated with EBR-II. The SCMS facility is used to remove sodium from reactor components for repair or replacement.

The TREAT reactor is an uranium oxide-fueled, graphite-moderated, air-cooled reactor. It was designed to produce short, controlled bursts of nuclear energy in order to simulate accident conditions leading to nuclear fuel damage. The reactor became operational in 1959. Tests at TREAT provide data on fuel cladding damage, fuel motion, coolant channel blockages, molten fuel/coolant interactions, and potential explosive forces during an accident.

ZPPR is the national facility for testing the physics properties of advanced, fast-spectrum reactors. ZPPR is designed to study the properties of experimental reactor cores. Experimental cores are built by hand-loading plates of reactor materials into drawers, which are then put into the designed pattern. The designs are tested at low power levels to determine characteristics of the core.

FCF (formerly called HFEF/S) became operational in 1964 and was used to demonstrate pyrometallurgical fuel reprocessing for EBR-II fuel during the first few years of operation. In

that mode of operation, a remotely operated production line was used for processing and refabricating spent EBR-II fuel and returning it to the reactor. After successful demonstration of this process in 1969, this mission was discontinued, and the facility was used to examine irradiated fuels and material experiments from EBR-II and TREAT and to provide other reactor support services such as spent fuel transfer to ICPP. FCF consists of two hot cells, one with an air atmosphere and the other with an inert argon-gas atmosphere. There are a total of 23 hot cell work stations around the outside perimeter of the FCF hot cells and four active work stations in the center work space of the argon cell. FCF is now being modified for use in demonstrating new remote recycling and refabrication fuel cycle processes for DOE. The facility has been upgraded and reequipped with new process equipment to carry out this demonstration.

HFEF (formerly HFEF/N) went into operation in 1975 and is used for examining irradiation experiments. Examinations conducted in the HFEF provide data that are essential for determining the performance and conditions of fuels and materials irradiated in the EBR-II reactor, the TREAT reactor, and other DOE reactor facilities. HFEF consists of two shielded hot cells: the decontamination cell, which contains an air atmosphere, and the main cell, which contains an argon gas atmosphere. Each of the 21 work stations in HFEF is equipped with shielded windows and master/slave manipulators. The main cell is used for work involving exposure of materials such as sodium, plutonium, and other materials that would react chemically with air.

A 250-kW TRIGA research reactor is located in the basement of HFEF and provides a source of neutrons for radiography. The Neutron Radiography Facility is equipped with two beam tubes and two separate radiography stations for neutron radiography of irradiated components. Facilities for decontaminating and repairing hot cell equipment and manipulators are also located within the HFEF.

The HFEF is currently being used for the examination and characterization of contact-handled TRU waste from RFP destined for the Waste Isolation Pilot Plant (WIPP).

The FMF contains the entire operation for the manufacturing of metallic fuel elements within a single building. The building contains a casting furnace and large gloveboxes for encapsulating and bonding the cast fuel slugs in a stainless-steel jacket.

Within the L&O Building is the analytical laboratory, which consists of hot cells, chemistry laboratories, and the Experimental Fuels Laboratory (EFL). The analytical laboratory provides chemistry support for ANL-W in the areas of environmental compliance, fuel chemistry, sodium/water chemistry, and waste classification analysis. The EFL is used in the development and fabrication of prototype metallic nuclear fuels.

The RLWTF receives low-level radioactive liquid waste from ANL-W facilities and stores the waste in storage tanks before evaporation in the shielded hot air drum evaporators (SHADEs). The L&O Building, FCF, and HFEF pipe liquid waste to the RLWTF facility directly. The RLWTF began operating in June 1983. Prior to this date, the low-level liquid evaporation process took place in the basement of the L&O Building.

During 1994–2003, it is expected that EBR-II and possibly other ANL-W facilities will be shut down. Some facilities will continue to operate, albeit with a revised mission related to waste research rather than fast reactor research. Activities related to the shutdown will take place over a period of several years.

Generation of the Waste. LLW generated at ANL-W for disposal at the RWMC SDA consists of contact-handled and remote-handled waste. Contact-handled LLW is generated in the FMF, L&O Building, FCF, HFEF, TREAT, ZPPR, EBR-II, SCMS, and RLWTF facilities. This waste is generated by routine facility operations, maintenance, monitoring, and modification. The waste includes filters, machinery components, scrap tools, SHADE units, incinerables and compactibles that exceed radiation limits for compactible waste, and metal items not suitable for sizing. Remote-handled LLW is generated in the FCF and HFEF. The majority of the remote-handled waste is generated from hot cell activities and includes obsolete or broken equipment, subassembly parts, highly contaminated rags and decontamination materials, various metal scrap, and glass.

The majority of the volume and radioactivity (curies) in ANL-W waste disposed of at the SDA in the past and forecasted for the future can be attributed to the FCF and HFEF waste streams. For the period 1984–1993, the FCF and HFEF remote-handled waste streams contributed the vast majority of the radioactivity in ANL-W waste. However, the volumes disposed of from the FCF and HFEF remote-handled waste streams made up less than 2% of the total volume of ANL-W waste disposed of. The FCF and HFEF contact-handled waste streams contributed approximately 72% of the total volume, but less than 1% of the total radioactivity.

During 1994–2003, the activities that generate the waste will change somewhat to reflect the shutdown activities. However, the quantity and nature of the waste are not expected to change significantly. About half of the total ANL-W waste volume sent to the SDA will be from the FCF contact-handled waste stream. Nearly all of the total activity in ANL-W waste sent to the SDA will be from the HFEF remote-handled waste stream.

General Availability of Information. The main sources of data pertaining to the waste from ANL-W in the time period 1984–1993 are the ANL-W shipping records and waste characterization reports used for approval of waste receipts at the RWMC.

Data available for the 1994–2003 projected waste includes the waste generator forecast of Grant (1993), and waste characterization reports for ongoing ANL-W waste streams.

Data-Collection Approach. The data collection approach used for the 1984–1993 period was to enter the ANL-W shipping record information into a spreadsheet/database and sort the data by shipment year, waste destination, generating facility, waste type, etc., to arrive at the total volumes of waste and total radioactivity from each facility, the waste container types, and the waste types. The radionuclide breakdowns for each facility were determined using the results of current waste sample analyses. Waste stream characteristics were gathered from ANL-W waste characterization reports.

The 1984–1993 total radioactivity for each shipment was assumed to be accurate in the shipping records. Distribution of radionuclide activity was modified to account for previous entries of MFP and MAP.

Data reported in the worksheets for waste expected to be shipped from ANL-W to the RWMC SDA for the time period 1994–2003 were gathered from the ANL-W radioactive waste forecast (e.g., Grant 1993) and waste characterization reports. Recent waste sample analysis results were used to determine the facility-specific radionuclide breakdowns.

Description of Waste Streams. The ANL-W waste sent to the SDA for the time period 1984–1993 is divided into eight waste streams (see Table 2-14). Some of these waste streams were extensions of the waste streams identified in the previous report on historical (1952–1983) waste burial.

The ANL-W waste projected to be sent to the SDA for the time period 1994–2003 is divided into 11 waste streams, eight of which are extensions of the waste streams for the 1984–1993 time period and three that are nonroutine, one-time waste streams (see Table 2-15).

2.5.6 Other Generators

The Generators. "Other" generators for the recent time period include one offsite generator, ANL-E (termed ALE in RWMIS), and seven onsite generators: the Auxiliary Reactor Area (ARA), CFA, PBF, RWMC, D&D, WERF, and Combustion Engineering—General Atomics (CEG). Some of these generators produced more than one waste stream, but the total radioactivity produced was very small, amounting to less than 0.1% of the total radioactivity received at the RWMC for burial in the SDA from all waste generators during 1984–1993.

The ALE is located near Chicago, Illinois. Waste was received from this facility beginning in 1980, and the last shipments were received in 1988. No additional waste is expected to be received at the INEL from this generator. Most of this waste was produced from research activities.

ARA is located in the south-central portion of the INEL. It was known as the Army Reactor Area until 1965, when the Army's programs at the INEL were phased out. Waste was produced during the time period 1984–1987 as the result of additional cleanup of the area, which had been subjected to D&D earlier. No waste was sent to the RWMC SDA from this facility after 1987.

CFA is also located in the south-central portion of the INEL. Some of the facilities in use at CFA were built in the 1940s and 1950s to support and house Naval Gunnery Range personnel. These facilities have been modified continually over the last 40 years to meet the changing needs of the INEL. CFA currently operates as a centralized location for support of the other INEL facilities, including administrative support, service shops, sanitary landfill, warehousing, security support, laboratory services, training, medical services, and receiving and storage. Small amounts of waste are produced from these operations.

Table 2-14. Waste streams generated at Argonne National Laboratory-West during 1984–1993.

Waste stream number	Description of waste
ANL-704-1R	Contact-handled, nonprocessible LLW generated during the manufacturing of metallic fuels and facility operations in the FMF and Fuel Assembly Storage Building facilities
ANL-752-1R	Contact-handled, nonprocessible LLW generated during L&O facility operations, maintenance, modifications, and monitoring
ANL-765-1R	Contact-handled, nonprocessible LLW generated during FCF operations, maintenance, modifications, and monitoring
ANL-765-2R	Remote-handled, subassembly LLW generated during nuclear fuel and materials experiments in the FCF
ANL-785-1R	Remote-handled, subassembly LLW generated during nuclear fuel and materials experiments in the HFEF
ANL-785-2R	Contact-handled, nonprocessible LLW generated during HFEF operations, maintenance, modifications, and monitoring
ANL-793-1R	Contact-handled, nonprocessible LLW generated during SCMS, EBR-II, TREAT, and ZPPR facility operations, maintenance, modifications, and monitoring
ANL-798-1R	Contact-handled, nonprocessible LLW generated during facility maintenance, monitoring, and the evaporation of low-level liquid waste in the SHADEs

Table 2-15. Waste steams expected to be generated at Argonne National Laboratory—West during 1994–2003.

Waste stream number	Description of waste
ANL-704-1P	Contact-handled, nonprocessible LLW generated during the manufacturing of metallic fuels and facility operations in the FMF
ANL-752-1P	Contact-handled, nonprocessible LLW generated during L&O facility operations, maintenance, modifications, and monitoring
ANL-763-1P	Contact-handled LLW consisting of sludge solidified with grout, soil, rocks, and concrete pieces generated during the cleanup of the EBR-II leach pit (not routine, generated in 1993, shipped in 1994, no future generation)
ANL-765-1P	Contact-handled, nonprocessible LLW generated during FCF operations, maintenance, modifications, and monitoring
ANL-765-2P	Remote-handled, subassembly LLW generated during nuclear fuel and materials experiments in the FCF
ANL-785-1P	Remote-handled, subassembly LLW generated during nuclear fuel and materials experiments in the HFEF
ANL-785-2P	Contact-handled, nonprocessible LLW generated during HFEF operations, maintenance, modifications, and monitoring
ANL-793-1P	Contact-handled, nonprocessible LLW generated during SCMS, EBR-II, TREAT, and ZPPR facility operations, maintenance, modifications, and monitoring
ANL-798-1P	Contact-handled, nonprocessible LLW generated during facility maintenance, monitoring, and the evaporation of low-level liquid waste in the SHADEs
ANL-INC-1P	Incinerable LLW from maintenance and facility operations (generated from 1992 through 1994, to be shipped to Scientific Ecology Group (SEG) for incineration in 1995 and the resulting ash to be shipped to the SDA in 1995)
ANL-COM-1P	Compactible LLW from maintenance and facility operations (to be shipped to WERF in 1995 for compaction and the resulting compacted waste to be shipped to the SDA in 1995)

The PBF area is located approximately 6 mi northeast of CFA. This area originally contained reactors constructed for the Special Power Excursion Reactor Test (SPERT) experiments. Four SPERT reactors were built beginning in the late 1950s as part of an early investigation involving reactor transient behavior tests and safety studies on water-moderated, enriched-fuel reactor systems. All of the reactors have been removed, and most of the facilities have undergone D&D. PBF presently consists of the PBF reactor area (north of SPERT-I), PBF control area, Waste Engineering Development Facility (at the SPERT-II site), WERF at the SPERT-III site, and Mixed Waste Storage Facility (SPERT-IV). Waste from WERF is reported separately from the PBF waste.

Small amounts of waste are generated by operations at the RWMC itself.

The D&D of INEL nuclear facilities has been in progress as a separate function since 1975. Areas subjected to D&D activities that produced waste during the time period 1984–1993 are the ARA, the Boiling Water Reactor Experiment V (BORAX V) area, the Army Reentry Vehicle Area (ARVFS), CFA, the LOFT area [now called the Containment Test Facility (CTF) area], the Heat Transfer Reactor area at TAN, the IET area at TAN, the MTR area at TRA, and the Power Excursion Reactor area (PER or SPERT).

WERF receives waste from all INEL facilities for treatment. Three treatment processes are involved: incineration, waste sizing, and compaction. The primary purpose of these treatments is to reduce the volume of the waste so that less space is required for burial in the SDA. The incineration process handles combustible waste, and the volume reduction averages about 200:1. The ash is disposed of in the SDA without further treatment if it passes the EPA toxicity leach test. If it does not pass this test, it is stabilized with Portland cement before disposal. The waste sizing process uses manually operated torches and mechanical cutting devices, including saws and shears. Both metallic and wood structures are sized. The average size reduction is 5:1. The compaction process uses a unit with a compaction force of 200 tons. The waste is compacted into metal containers for disposal in the SDA. The average size reduction for this process is 5:1.

At present, only waste sizing and compaction are in operation at WERF. Waste incineration is shut down and is expected to restart in 1995.

The other generators for the time period 1994–2003 include five of the generators from 1984–1993 (ARA, CFA, D&D, WERF, and RWMC), and one new generator [environmental restoration projects (ERP)]. The total activity from these streams is projected to be very small compared with the total activity projected for disposal at the SDA during 1994–2003.

Generation of the Waste. Waste produced by the other generators includes scrap metals, combustible materials slightly contaminated with radionuclides, and a variety of wastes associated with research and development and the cleanup of facilities.

Waste from ARA consists primarily of cleanup waste resulting from the past short-term operation of the Army Gas-Cooled Reactor Experiment (GCRE), the Mobile Low-Power Reactor No. 1 (ML-1), the Stationary Low-Power Reactor No. 1 (SL-1), and the radiochemistry laboratory.

Waste from CFA is from several facilities, including the CFA laundry, machine shops, maintenance shops, sewage treatment facilities, and the Radiological and Environmental Sciences Laboratory (RESL). It includes combustibles and biological waste.

Waste generated by D&D operations consists primarily of surface-contaminated metal, concrete, and soil resulting from the dismantling and decommissioning of buildings and building components.

The PBF area radioactive waste was produced from earlier operations of the four SPERT reactors. This waste is primarily metals and combustibles.

Virtually all of the waste shipped from WERF to the SDA originated at other INEL facilities. WERF produced (a) ash by incinerating combustible materials, (b) metals from the size reduction of large pieces of metal equipment by sawing and cutting, and (c) compacted waste by compacting large pieces of metal and wood in a compactor.

Waste was produced at the RWMC itself, usually by decontamination of equipment.

The offsite generator, ALE, produced waste primarily as a result of conducting research in physical, biomedical, and environmental sciences that involved the use of radioactive materials. Some of the waste arose from performing modifications to existing research facilities.

The waste generator CEG processed a lithium target prepared and irradiated at TRA. The target was sent offsite to CEG for processing, but the waste was returned to the INEL for burial at the SDA because the tritium had been generated at TRA. This was a one-time experiment associated with research for the New Production Reactor (NPR) project, which has been canceled.

The waste produced by the other generators for burial in the SDA during 1994–2003 is expected to be similar to the waste produced during 1984–1993. The radioactivity is expected to be small, and the volumes are expected to be similar or smaller. Two exceptions are that no shipments of waste from the offsite generator ANL-E are expected, and no shipments are expected from CEG. CEG was a one-time generator for NPR experiments. Waste from PBF is projected to contain insignificant amounts of activity in comparison to the other generators. This assumes that PBF remains shut down with no plans to inactivate the reactor.

General Availability of Information. The main sources of information available for the other generators for the period 1984–1993 were technical notes produced from information in RWMIS, RWMIS itself, and topical reports. Numerous reports were available on the D&D projects: Moser (1986), Schoonen (1984), Kaiser (1984), Suckel (1984), Smith (1984), Smith (1985a), Smith and Scott (1984), Bradford (1984), Smith (1985b), Stoll (1987), Browder and Wills (1985), McCusker (1989), Arave and Rodman (1992), Evans and Perry (1993), Moser (1984), Moser (1993), Hansen (1993), Dolenc (1986), and Rhoades (1988). These reports are useful for determining if any hazardous materials were disposed of in the SDA, since only radioactive materials are recorded in RWMIS. Individual waste shipment records were not available, but the RWMIS records generally contained considerable detail pertaining to the activities of the individual radionuclides in each waste shipment.

Projections of waste to be shipped from other generators to the RWMC in the period 1994–2003 were available from several radioactive waste forecasts: Hitz and Skinner (1993), Keating (1993), Kaiser (1993), Leonard (1994), Banister (1993), Gibbon (1993), Ellison et al. (1993), Miley and Bishoff (1993), Luptak (1993), Miller (1993), and Sayer and Lewis-Kido (1993). Many of the individuals who prepared these forecasts were contacted to clarify specific points. Other documents provided useful information for projecting future waste: EG&G Idaho (1993), Klassy and Keller (1994), Buckland et al. (1993), and Maheras et al. (1994).

Data-Collection Approach. The approach used to obtain data for the other generators for the 1984–1993 period involved using the technical notes for each waste generator. This source provided the volume, total number of curies, radionuclide distribution, and type of container. The content codes associated with the waste for a particular waste generator were then obtained from the RWMIS records, and a description of the waste was determined from these codes. Several reports, as listed in the previous section, were also reviewed to obtain information.

The approach used to obtain data for the other generators for the 1994–2003 period was to review the radioactive waste forecasts for this period to obtain the necessary information, and to contact the individuals who prepared the forecasts for further clarification of specific points.

Description of Waste Streams. The waste generated by the other facilities in the period 1984–1993 was divided into a total of 15 streams (see Table 2-16).

The waste generated by the other facilities in the period 1994–2003 was divided into a total of eight streams (see Table 2-17).

2.6 Data Qualification Process

As shown in Figure 2-2, after the waste information for each generator was collected and entered onto data forms (one form for each waste stream), the information was subjected to a qualification process. That process is described briefly here.

Completed draft forms were logged in at a central point, and copies were reviewed by the technical lead personnel for the task. The completed draft forms were reviewed on points such as completeness, clarity, consistency, reasonableness of assumptions, use of appropriate scientific units, possible duplication or overlap of coverage with forms completed for other waste streams, and compatibility with the structure of the database.

The review comments were discussed with the data gatherer who had prepared the draft forms. After agreement was reached on resolution of the comments, the original preparer made any necessary revisions to the forms.

The forms were then logged in again at the central point and transmitted to the database personnel for entry. All data entered into the database (discussed in Section 2.7) were independently checked for correct entry. During data entry, several validation tables were used to ensure that only valid information was entered into several data fields. The validation tables

Table 2-16. Waste streams generated by other generators during 1984–1993.

Waste stream number	Description of waste
ARA-626-2R	Waste from cleanup of the hot cell, including combustible waste such as paper, cloth, plastic, and wood
ALE-317-1R	Primarily biological waste, such as tissue, feces, and other waste from small animals such as dogs and mice
ALE-317-2R	Cloth, paper, wood, plastic, cut-up scrap, cut-up glove boxes, and undefined general plant waste
ALE-317-3R	HEPA and other types of filters
ALE-317-4R	Construction materials from modifications of buildings and repair of roads, including paving and roofing materials, concrete, soil, and brick
ALE-317-5R	Liquids absorbed in dry materials such as vermiculite; also, sludge and evaporator bottoms absorbed in dry solids
ALE-317-6R	Metals
CEG-CEG-1R	A powder produced by crushing small silicon carbide particles containing lithium target material and solidified in Aquaset; this waste originated as part of the NPR project and was a "one time" waste
CFA-617-2R	Liquids absorbed in vermiculite, plastic, paper, cloth, carbon steel, wood, sludge, and some biological waste
D&D-ALL-1R	Piping, valves, evaporators, pumps, concrete, rags, paper, HEPA filters, etc., from D&D at 11 locations
PBF-620-1R	Primarily waste associated with cleanup, including metal tanks, ion-exchange resin, concrete blocks, glass, paper, wood, cloth, etc.
WER-COM-1R	Primarily metal and wood that has been reduced in volume by compacting
WER-INC-1R	Ash from the incineration of combustible materials, e.g., paper, wood
WER-MET-1R	Primarily metals that have been cut into smaller pieces
WMC-WMC-2R	Miscellaneous waste, usually from decontamination of equipment

Table 2-17. Waste streams expected to be generated by other generators during 1994–2003.

Waste stream number	Description of waste
ARA-ALL-1P	Generally pieces of metal, concrete, and rubble resulting from the removal of additional structures at ARA-I, ARA-II, and ARA-III
CFA-625-1P	Principally protective clothing and personal protective equipment
D&D-ARV-1P	Concrete, steel, lead, and NaK (after it is treated) from the disassembly and disposal of the ARVFS underground bunker and its contents (stored NaK)
ERP-ALL-1P	Primarily soil, sludge, and other debris resulting from the remediation of contaminated sites
WER-COM-1P	Noncombustible materials, such as light-gauge metal, glass, and acid-producing combustible materials, such as halogen-containing plastics that exceed allowable radiation limits for incineration after compaction in WERF
WER-INC-1P	Ash from the incineration at WERF of combustible materials from all of the INEL facilities
WER-MET-1P	Metal from all of the INEL facilities that has been reduced in size at WERF by cutting and sawing
WMC-ALL-1P	Cleanup materials and miscellaneous waste

contain prespecified "acceptable" values for the following types of information (data fields): nuclide, chemical name, Chemical Abstract Services (CAS) number, generator, building, etc. As a final check, the database printouts were then reviewed by the data gatherers who had completed the original forms.

The information in this report, including the waste inventory printouts, underwent peer review by technical, program management, regulatory compliance, and waste generator personnel.

2.7 Contaminant Inventory Database for Risk Assessment

A convenient method was needed to use the large body of data captured on the data forms provided by the data gatherers for the preceding companion study (LITCO 1995) and the present study. Therefore, a database, called the Contaminant Inventory Database for Risk Assessment (CIDRA), was created to manage the data from both studies.

All data contained in CIDRA originated from completed data forms, a blank version of which can be found in Appendix A.

The CIDRA application was created in FOXPRO and is accessible in dBASE.

Textual information captured in the database can be aggregated over different fields in the database (e.g., by waste stream or by generator). However, query and sort capabilities on the text fields are limited. This information was electronically captured to maintain a record of how the waste stream information was obtained and other pertinent details about the waste stream. The data tracking form is hierarchical in that each subsequent section of the form contains more detailed information about a waste stream inventory.

The CIDRA report software application was developed to support reporting. The application can produce the following set of standard reports:

- Hazardous chemicals (Part C) data by various groupings [e.g., waste stream, generator(s)]
- Radionuclides (Part D) information by various groupings [e.g., waste stream, generator(s)].

The information in these reports consists of the quantities and respective units of radiological and nonradiological contaminants.

Report generation is augmented by an algorithm that was developed to perform simplified decay calculations on the radionuclides. The user may specify any date to which decay is calculated, and a data set with the decayed quantities is produced.

References for Section 2

Adams, J., 1985, *PBF-CANDU Fuel Element Loss-of-Coolant Accident Experiment Test Results Report*, EGG-2384, EG&G Idaho, Inc., May 1985.

Adams, J. P., J. C. Birchley, N. Newman, E. W. Coryell, M. L. Carboneau, S. Guntry, and L. T. Siefken, 1985, *Quick-Look Report on OECD LOFT Experiment LP-FP-2*, OECD LOFT-T-3804, EG&G Idaho, Inc., September 1985.

Arave, A. E., and G. R. Rodman, 1992, *Final Report of the Decontamination and Decommissioning of the BORAX-V Facility Turbine Building*, EGG-2683, EG&G Idaho, Inc., December 1992.

Banister, M. W., 1993, "Solid Radioactive Waste Forecast for TAN," EG&G Idaho, Inc., November 18, 1993.

Bartolomucci, J. A., 1989, letter to J. N. Davis, "Curie Content Estimates for ECF Scrap Casks," NRFE-E-1448, Naval Reactors Facility, February 27, 1989.

Bradford, J. D., 1984, *Final Report-Decommissioning of the SPERT-III Large Leach Pond at the Idaho National Engineering Laboratory*, EGG-2306, EGG-2275, EG&G Idaho, Inc., April 1984.

Browder, J. H., and E. L. Wills, 1985, *Decommissioning of the MTR-605 Process Water Building at the Idaho National Engineering Laboratory*, EGG-2361, EG&G Idaho, Inc., January 1985.

Buckland, R. J., D. J. Kenoyer, and D. M. Preusner, 1993, *INEL D&D Long-Range Plan*, EGG-WM-10924, EG&G Idaho, Inc., October 1993.

Carboneau, M. L., R. L. Nitschke, D. C. Mecham, E. W. Coryell, and J. A. Bagues, 1987, *OECD LOFT Fission Product Experiment LP-FP-2 Data Report*, OECD LOFT-T-3805, EG&G Idaho, Inc., May 1987.

Charter, J. L., 1993, letter to K. A. Taylor, "Radioactive Waste Forecasts - Naval Reactors Facility," Letter NRFRC-RCE-2247, November 15, 1993.

DOE, 1993, *Waste Stream Projections for Environmental Restoration at the Idaho National Engineering Laboratory*, DOE/ID-10417, Revision 2, U.S. Department of Energy, December 1993.

DOE, 1994a, *Waste Management Program Strategic Plan*, DOE/ID-10429, Revision 1, U.S. Department of Energy, June 1994.

DOE 1994b, *Draft Environmental Assessment: Test Area North Pool Stabilization Project*, draft, U.S. Department of Energy, February 1995.

Dolenc, M. R., 1986, *Final Report-Decontamination and Decommissioning of the SPERT-I Reactor Building at the Idaho National Engineering Laboratory*, EGG-2399, EG&G Idaho, Inc., February, 1986.

EG&G Idaho, 1988, *Waste Management Plan, Annual Update*, Revision 2, December 1988.

EG&G Idaho, 1993, *Waste Experimental Reduction Facility Safety Analysis Report*, Volume I, WM-F1-810026, Revision 9, EG&G Idaho, Inc., June 1993.

Ellison, J. J., C. M. Hiaring, and W. Malone, 1993, "Solid Radioactive Waste Forecast," EG&G Idaho, Inc., November 5, 1993.

EPRI, 1987, *Updated Scaling Factors in Low Level Radwaste*, EPRI NP-5077, Impell Corporation, March 1987.

Evans, J. C., E. L. Lepel, R. W. Sanders, D. L. Wilkerson, W. Silker, C. W. Thomas, K. H. Abel, and D. R. Robertson, 1984, *Long-Lived Activation Products in Reactor Materials*, NUREG/CR-3474, Battelle Pacific Northwest Laboratory, August 1984.

Evans, T. A., and E. F. Perry, 1993, *Final Report of the Remedial Action Taken for the TAN Building 616 Liquid Waste Treatment Facility*, EGG-2714, EG&G Idaho, Inc., September 1993.

General Electric Company, 1989, *Nuclides and Isotopes*, 14th edition, 1989.

Gibbon, D. O., 1993, "Solid Radioactive Waste Forecast for TAN," EG&G Idaho, Inc., November 16, 1993.

Graff, A. G., 1980, *ORIGEN2—A Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code*, ORNL-5G21, Oak Ridge National Laboratory, July 1980.

Grant, R. P., 1993, letter to V. C. Randall, "Radioactive Waste Forecast for 1994," Argonne National Laboratory—West, December 20, 1993.

Hansen, C. B., 1993, *Final Report for the Decontamination and Decommissioning of the SPERT-IV Waste Holdup Tank Ancillaries*, EGG-ER-11000, EG&G Idaho, Inc., September 1993.

Harker, Y. D., and D. W. Akers, 1994, *TRA Activity Weighting Factors/Physical and Chemical Properties of ^{14}C, ^{99}Tc, and ^{129}I*, ER-WAG7-51, engineering design file, EG&G Idaho, Inc., October 1994.

Harker, Y. D., 1995a, *Scaling Factors for Waste Activities Measured by G-M Method*, EDF-ER-WAG7-57, engineering design file, Lockheed Idaho Technologies Company, February 1995.

Harker, Y. D., 1995b, letter to T. H. Smith, "I-129 Act. in TRA Sections of HDT and RPDT Reports," YDH-2T-95, Lockheed Idaho Technologies Company, April 1995.

Hitz, J. R., and R. L. Skinner, 1993, letter to K. A Taylor, "Solid Radioactive Waste Forecast," RLS-018-93, Westinghouse Idaho Nuclear Company, Inc., November 15, 1993.

Hoff, D. L., R. G. Mitchell, R. Moore, and L. Bingham, 1993, *The Idaho National Engineering Laboratory Site Environmental Report for Calendar Year 1992*, DOE/ID-12082(93), U.S. Department of Energy, June 1993.

Kaiser, L. L., 1984, *Decontamination and Decommissioning MTR-657 Plug Storage Facility*, EGG-2286, EG&G Idaho, Inc., January 1984.

Kaiser, K. A, 1993, "Solid Radioactive Waste Forecasts," EG&G Idaho, Inc., November 15, 1993.

Keating, S. J., 1993, letter to V. C. Randall, "Test Reactor Area (TRA) Radioactive Waste Forecasts," SJK-005-93, EG&G Idaho, Inc., November 17, 1993.

Klassy, C. E., and D. E. Keller, 1994, *Auditable Safety Analysis for the Storage and Handling of the EBR-I NaK*, EGG-ER-11160, Revision 0, EG&G Idaho, Inc., March 1994.

Klossner, K. A., 1993, letter to K. A. Taylor, "Solid Radioactive Waste Forecast," KAK-16-93, Babcock & Wilcox Idaho, Inc., November 15, 1993.

Leonard, P. R., 1994, letter to K. A. Taylor, "NRF Forecast for Waste Sent to WERF," PRL-15-94, EG&G Idaho, Inc., August 1994.

LITCO (Lockheed Idaho Technologies Company), 1995, *A Comprehensive Inventory of Radiological and Nonradiological Contaminants in Waste Buried in the Subsurface Disposal Area of the INEL RWMC During the Years 1952-1983*, INEL-95/0310, Rev. 1, formerly EGG-WM-10903, August 1995.

Litteer, D. L., 1988, *Idaho National Engineering Laboratory Radioactive Waste Management Information System (RWMIS) Solid Data Users Manual*, DOE/ID-10188, Revision 2, U.S. Department of Energy, September 1988.

Luptak, A. L., 1993, "Solid Radioactive Waste Forecast," EG&G Idaho, Inc., November 15, 1993.

Maheras, S. J., A. S. Rood, S. O. Magnuson, M. E. Sussman, and R. N. Bhatt, 1994, *Radioactive Waste Management Complex Low-Level Waste Radiological Performance Assessment*, EGG-WM-8773, EG&G Idaho, Inc., May 1994.

McCusker, T. K., 1989, *Final Report: Decontamination and Decommissioning of Heat Transfer Reactor Experiment Test Assemblies HTRE-2 and HTRE-3*, EGG-2575, EG&G Idaho, Inc., September 1989.

Miley, D. M., and J. R. Bishoff, 1993, "Solid Radioactive Waste Forecast," EG&G Idaho, Inc., October 18, 1993.

Miller, D. L., 1993, "Solid Radioactive Waste Forecast," EG&G Idaho, Inc., November 8, 1993.

Moser, C. L., 1993, *Final Report-CPP-603 Chloride Removal System Decontamination and Decommissioning*, WINCO-1124, Westinghouse Idaho Nuclear Company, Inc., February 1993.

Moser, C. L., 1984, *Final Report-ICPP BIF Filter Room Decontamination and Decommissioning*, WINCO-1028, Westinghouse Idaho Nuclear Company, Inc., December 1984.

Moser, C. L., 1986, *Final Report-Decontamination and Decommissioning of RALA Off-gas Cell and Storage Tank*, WINCO-1036, Westinghouse Idaho Nuclear Company, Inc., February 1986.

Mullen, C. K., 1994, letter to H. K. Peterson, "Transmittal of Memories of Test Area North (TAN) Wastes," CKM-01-94, EG&G Idaho, August 29, 1994.

NCRPM, 1987, *Exposure of the Population in the United States and Canada from Natural Background Radiation*, Report No. 94, National Council on Radiation Protection and Measurements, December 1987.

Nieslanik, R. W., 1994, letter to T. H. Smith, "NRF Comments to the Radioactive Waste Management Complex (RWMC) Waste Inventory Report," NRFEM-RR-1122, Naval Reactors Facility, March 29, 1994.

Randall, V. C., 1994, letter to J. A. Roche, "Transmittal of 1994 and 10-Year Radioactive Waste Forecasts," VCR-03-94, EG&G Idaho, Inc., January 24, 1994.

Rhoades, W. A., 1988, *Final Report-Decontamination and Decommissioning of the Auxiliary Reactor Area IV Facility*, EGG-2518, EG&G Idaho, Inc., March 1988.

Sayer, R. D., and G. Lewis-Kido, 1993, "Solid Radioactive Waste Forecast," EG&G Idaho, Inc., October 6, 1993.

Schnitzler, G. B., 1994, letter to E. B. Nieschmidt, "Radioisotopes in ATR Fuel Elements," BGS-2-94, EG&G Idaho, Inc., February 10, 1994.

Schoonen, D. H., 1984, *Reactor Vessel Decommissioning Project*, EGG-2298, EG&G Idaho, Inc., September 1984.

Smith, D. L., 1984, *Decontamination and Decommissioning of the TAN/TSF-3 Concrete Pad*, EGG-2292, EG&G Idaho, Inc., April 1984.

Smith, D. L., and J. G. Scott, 1984, *Final Report-Decontamination and Decommissioning of CPP-601 Process Cells "A," "B," "C," "D," and "L,"* EGG-2304, EG&G Idaho, Inc., September 1984.

Smith, D. L., 1985a, *Decontamination and Decommissioning of the BORAX-V Leach Pond*, EGG-2300, EG&G Idaho, Inc., January 1985.

Smith, D. L., 1985b, *Final Report Decontamination and Decommissioning of MTR-603 HB-2 Cubicle*, EGG-2431, EG&G Idaho, Inc., December 1985.

Stoll, F. E., 1987, *Final Report of the Decontamination and Decommissioning of the Initial Engine Test Facility and the IET Two-Inch Hot-Waste Line*, EGG-2468, EG&G Idaho, Inc., April 1987.

Suckel, R. A., 1984, *Final Report-Decontamination and Decommissioning of the SPERT-I Seepage Pit at the Idaho National Engineering Laboratory*, EGG-2291, EG&G Idaho, Inc., November 1984.

Weidner, J. R., 1993, letter to T. H. Smith, "Corrosion Properties of Stainless Steel at the RWMC and the Potential Release of Nickel and Chromium into the Environment," JRW-13-93, EG&G Idaho, August 12, 1993.

Wilks, P. H., 1962, *Comprehensive Technical Report, General Electric Direct-Air-Cycle Aircraft Nuclear Propulsion Program, Nuclear Safety*, APEX-921, General Electric Company, 1962.